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APPLIED HEALTH PHYSICS AND SAFETY

ANNUAL REPORT FOR 1968

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

550

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HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1968

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JULY 1969

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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UNION CARBIDE CORPORATION
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U. S. ATOMIC ENERGY COMMISSION

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3.0 CONTRIBUTIONS

The data for this report were contributed by: H. H. Abee, Environs Radiation Monitoring Section; R. L. Clark, Radiation and Safety Surveys Section; E. D. Gupton, Applied Radiation Dosimetry Section; A. D. Warden, Health Physics and Safety Associate Department Head; D. C. Gary, Industrial Engineering.

4.0 SUMMARY

There were no accidental releases of gaseous or liquid waste from the Laboratory which were of a reportable level as defined in AEC Manual Chapter 0524. The concentration of radioactive materials in the environs was well below the maximum levels recommended by the ICRP, FRC, and AEC.

No employee received an external or internal radiation dose of a level that required a report to the AEC. The highest whole body dose received by an employee was about 4.7 rem or 39 percent of the maximum permissible annual dose. There were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half of a maximum permissible body burden.

There were 20 unusual occurrences recorded during 1968, which is the second lowest number recorded since the present system of reporting unusual occurrences was established in 1960. The lowest number for any one year was 16, the number reported for 1967. The average number reported for the past five years was 26.

The ORNL safety record for 1968 was the best in the history of the Laboratory. There was only one Disabling Injury reported during the year, and there was a decrease in number of both Serious Injuries and Medical Treatment Cases as compared with 1967. The Disabling Injury Frequency Rate for 1968 was 0.13, as compared with the average rate of 1.19 for the previous five years, 1963-1967.

5.0 ENVIRONS MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of 22 stations which are positioned in relation to ORNL operational activities (Figures 5.1 and 5.2); the perimeter air monitoring (PAM) network consists of nine stations which are located on the perimeter of the AEC controlled area (Figure 5.3); and the remote air monitoring (RAM) network consists of eight stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL (Figure 5.4). The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, and (3) rainwater for measurement of fallout occurring as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fallout; and the rainwater data provide information on the soluble and insoluble fractions of the radioactive content of fallout material.[1]

Low-level radioactive liquid waste originating from ORNL operations are discharged, after preliminary treatment, to White Oak Creek, which is a small tributary of the Clinch River. Liquid waste releases are controlled so that the resulting average radioactive concentrations in the Clinch River are well below the maximum permissible concentrations for waste released to uncontrolled areas as recommended in Annex 1, Table II, of AEC Manual Chapter 0524.

The radioactive content of the White Oak Creek discharge is determined at White Oak Dam (Figure 5.5) which is the last control point along the stream prior to entry of White Oak Creek waters into Clinch River waters. Water samples are collected also at a number of locations along the Clinch River, beginning at a point above the entry of waste into the river via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the confluence of White Oak Creek and the Clinch River. Water samples are analyzed for gross radioactivity and for certain specified long-lived radionuclides. A weighted average maximum permissible concentration, $(MPC)_w$, for the mixture of radionuclides is calculated on the basis of the isotopic distribution in the water.

Samples of ORNL potable water are collected daily, composited and stored. At the end of each quarter these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma emitting radionuclides by gamma spectrometry.

Raw milk samples are collected at 12 sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from eight stations which

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[1] A detailed discussion concerning techniques used in processing air and water samples for environmental monitoring purposes is given in ORNL-2601, <u>Radioactive Waste Management at Oak Ridge National Laboratory</u>.

are located outside the AEC controlled area within a 12-mile radius of ORNL (Figure 5.6). Samples are collected every five weeks from the four remaining stations, all of which are located outside the 12-mile radius up to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of waste releases originating from ORNL operations; second, samples collected remotely to the immediate vicinity of the ORNL area provide background data which are essential in establishing a proper index from which the intentional or accidental release of radioactive materials originating from Oak Ridge operations may be evaluated.

Aerial background surveys are made over the ORNL area and for several miles from ORNL in the general direction of low altitude prevailing winds. The frequency of flights has been established at once per quarter.

Background gamma radiation measurements are made monthly at a number of locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

River bottom sediments in the Clinch and Tennessee Rivers have been surveyed and analyzed annually since the year 1951 for the purpose of providing data relative to the dispersion of radioactive waste released from Oak Ridge operations to the Clinch River. This survey is normally carried out with the help of personnel from the summer student program. In keeping with Laboratory policy to reduce operating costs during fiscal year 1969, no summer participants were hired and no river survey was performed during the summer of 1968.

Fish from the Clinch River are sampled during the spring and summer each year and analyzed for their radioactive content. The radionuclide concentration in fish are related quantitatively to potential human intake of radioactivity through consumption of fish.

5.1 Atmospheric Monitoring

5.1.1 Air Concentrations - The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM networks during 1968, were as follows:

Network	Concentration (μCi/cc)
LAM	0.29×10^{-12}
PAM	0.16×10^{-12}
RAM	0.16×10^{-12}

The LAM network value of 0.29×10^{-12} µCi/cc is about 0.01 percent of the $(MPCU)_a$[2] based on occupational exposure of 3×10^{-9} µCi/cc. Both the PAM and RAM network values represent ~0.2 percent of the $(MPCU)_a$ of 1×10^{-10} µCi/cc applicable to waste released to uncontrolled areas. A tabulation of data for each station in each network is given in Table 5.1. The weekly values for each network are illustrated in Table 5.2.

The number of radioactive particles collected on the air monitoring filters is shown in Table 5.1. Data are given on both the activity range of the particles and to the total number of particles per 1000 cu. ft. of air sampled.

5.1.2 Fallout (Gummed Paper Technique) - Radioparticulate fallout as measured by both the LAM and RAM networks decreased by a factor of 16 from the values measured in 1967. The value measured by the PAM network of stations decreased by a factor of 21 from the 1967 value. Table 5.3 gives the network averages by weeks. Table 5.4 gives a tabulation of data for each station within each network.

5.1.3 Atmospheric Radioiodine (Charcoal Collector Techniques) - Atmospheric radioiodine measured by the perimeter stations averaged 0.013×10^{-12} µCi/cc during 1968. This is only about 0.01 percent of the maximum permissible concentration of 1.0×10^{-10} µCi/cc applicable to waste released to uncontrolled areas. The maximum value observed at any one station for one week was 0.29×10^{-12} µCi/cc. This value was measured at PAM 34, the perimeter station nearest the Plant area. Table 5.5 compares the weekly discharge of radioiodine from ORNL stacks[3] with the average concentration of radioiodine measured by the perimeter stations.

The average radioiodine concentration measured by the local stations was 0.13×10^{-12} µCi/cc. This is about 0.01 percent of the maximum permissible concentration for occupational exposure. The maximum value observed on any one station for one week was 2.6×10^{-12} µCi/cc. This value was observed at LAM 4 (near the waste treatment plant). Table 5.6 gives ^{131}I data for both the Plant area (LAM's) and the perimeter area monitors.

5.2 Water Analyses

5.2.1 Rainwater - The average concentration of radioactivity in rainwater collected from the three networks during 1968 were as follows:

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[2] The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. AEC Manual Chapter 0524, Appendix, Annex 1, gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3] "Summary of Waste Discharges", Weekly Reports, 1968, L. C. Lasher.

Network	Concentration (μCi/ml)
LAM	0.27×10^{-7}
PAM	0.34×10^{-7}
RAM	0.33×10^{-7}

The value observed on the LAM network was essentially the same as that measured during 1967. The average values for the PAM and RAM networks are not significantly different from the average value for the LAM network. The average values for each station are shown in Table 5.7; the average values for each network for each week are given in Table 5.8.

5.2.2 Clinch River Water - A total of 16 beta curies of radioactivity was released to the Clinch River during 1968 as compared to 40 for 1967 (Table 5.9). Yearly discharges of radionuclides to Clinch River, 1949 through 1968, are shown in Table 5.10. Radiochemical analysis of the White Oak Dam effluent indicated that about 31 percent of the radioactivity was ^{106}Ru. The percentage of ^{90}Sr in the effluent was 17.5 in 1968 compared to 13 in 1967.

The calculated average concentration of radioactive materials in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the River) was 1.0×10^{-8} μCi/ml. This represents only 0.83 percent of the weighted average $(MPC)_w$ for waste released to uncontrolled areas (Table 5.11). The average concentration of radioactive materials in the Clinch River did not exceed 4.5 percent of the $(MPC)_w$ during any week in 1968 (Table 5.12).

The measured average concentration of radioactivity in Clinch River water at CRM 23.1 (above the entry of White Oak Creek) was 0.17 percent of the weighted average $(MPC)_w$ (Table 5.11). The concentration of ^{90}Sr in the River above the entry of White Oak Creek was about two-thirds of the contribution calculated for White Oak Creek effluent at CRM 20.8 assuming uniform mixing of the two streams.

A significant concentration of ^{60}Co (Table 5.11) was detected in Clinch River water at CRM 23.1 (upstream from the Laboratory waste outfall) during the fourth quarter of 1968. The contamination came from sources other than Laboratory operations (see Section 5.2.3 Potable Water).

The measured average concentration of radioactive materials in the Clinch River at CRM 4.5 (near Kingston, Tennessee) was 0.69×10^{-8} μCi/ml. This value represents 0.52 percent of the $(MPC)_w$ applicable to waste released to uncontrolled areas.

5.2.3 Potable Water - The average concentrations of ^{90}Sr in potable water at ORNL during 1968 were as follows:

Quarter Number	Concentration ^{90}Sr ($\mu Ci/ml$)
1	0.45×10^{-9}
2	0.14×10^{-9}
3	0.45×10^{-9}
4	0.36×10^{-9}
Average for Year	0.35×10^{-9}

The average value of 0.35×10^{-9} represents 0.12 percent of the $(MPC)_w$ for drinking water applicable to individuals in the general population.

Cobalt-60 was detected in the potable water at ORNL during the fourth quarter of 1968. The composite sample for this period showed a ^{60}Co content of 0.026 d/m/ml. This value is $\sim$0.03 percent of $(MPC)_w$ for soluble ^{60}Co for application to uncontrolled areas.

The source of this contamination was found to be a contaminated branch (Braden Branch) that flows into Melton Hill Lake upstream from the intake of the Oak Ridge city (ORNL potable waster source) water supply. The ^{60}Co content of this branch was, when sampled in February, 1969, 67 d/m/ml or $\sim$60 percent of $(MPC)_w$ for ^{60}Co.

5.3 Milk Analyses

The average concentration of ^{90}Sr in raw milk samples collected within a 12-mile radius of the Laboratory during 1968 was 20 pCi/l. The average concentration of ^{90}Sr in samples collected between 12 miles and 50 miles from the Laboratory was 19 pCi/l. These results would indicate that the ^{90}Sr content of milk in the Oak Ridge area is from sources other than the Laboratory. Table 5.13 presents the weekly average concentration of ^{90}Sr in raw milk collected from the immediate environs of Oak Ridge.

The average concentration of ^{131}I in raw milk samples collected within a 12-mile radius of the Laboratory, as well as the samples collected between 12 miles and 50 miles from the Laboratory, was below the minimum detectable level of 10 pCi/l, except for the fourth, fifth and sixth weeks of the year. The highest average for any one week, the sixth week, was 18.6 pCi/l. (When levels are below 10 pCi/l, for averaging purposes a value of 5 pCi/l is assumed.) Table 5.5 includes the weekly average concentration of ^{131}I in raw milk collected at the stations within a 12-mile radius of the Laboratory and the weekly discharge of ^{131}I from the ORNL stacks.

The average yearly values for both ^{90}Sr and ^{131}I fall within the limits of FRC Range I daily intake guides, if one assumes an intake of 1 liter of milk per day.

5.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each location on a frequency of once each five weeks. The average background level during 1968 as measured at these stations was 0.011 mR/hr. Average background readings and the location of each station are presented in Figure 5.7.

Background measurements made monthly with a calibrated GM monitor at five selected locations adjacent to the ORNL area yielded an average background reading of 0.011 mR/hr during 1968. Corresponding measurements made at 53 locations on the ORNL site gave an average background of 0.076 mR/hr. The average background level measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite Reactor was 0.012 mR/hr. A comparison of average background values taken both on and off the X-10 site for the years 1957-68 is presented in Figure 5.8.

5.5 Radionuclides in Clinch River Fish

Three species of fish were collected from the Clinch River for assay during the spring and summer of 1968. The fish were prepared for radiochemical analysis in a manner analogous to human utilization. Ten fish of each species were composited for each sample and analyzed by gamma spectrometry and radiochemistry for the critical radionuclides contributing most heavily to potential radiation dose to man.

The concentration of radionuclides found in Clinch River fish during the years 1965, 1966, 1967 and 1968 are given in Table 5.14. Smallmouth buffalo are considered to be representative of the commercial fish species and the white crappie is the most commonly caught game fish. Gizzard shad were included because of the potential utilization of this species for fish meal or cat and dog food.

The data in Table 5.14 show some of the difficulties encountered when attempting to utilize one sample a year for interpretation of results. There is sufficient variation in results to make detailed explanation rather tenuous. Within five to ten years, however, the results should reflect the general trend in activity releases to the river and the full value of the analyses may then be realized.

An estimate of man's intake of radionuclides from eating Clinch River fish was made by assuming an annual rate of fish consumption of 37 lbs.[4] A maximum permissible intake (MPI) was calculated by assuming a daily intake of 2.2 l of water containing the MPC_w of the radionuclide in question. The fraction of MPI attained for each radionuclide of interest was calculated from the estimated intake of contaminated fish, Table 5.15.

[4] "Evaluation of Radiation Dose to Man from Radionuclides Released to the Clinch River", K. E. Cowser, W. S. Snyder, C. P. McCammon, C. P. Straub, O. W. Kochtitzky, R. L. Hervin, E. G. Struxness, and R. J. Morton.

Table 5.1 Concentration of Radioactive Materials in Air--1968
(Filter Paper Data--Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μCi/cc	No. of Particles by Activity Ranges $< 10^5$ d/24 hr	10^5–10^6 d/24 hr	10^6–10^7 d/24 hr	$> 10^7$ d/24 hr	Total	Particles Per 1000 ft^3
			Laboratory Area					
HP-1	S 3587	2.5	3.3	0.06	0.00	0.00	3.4	0.14
HP-2	NE 3025	2.9	1.2	0.00	0.00	0.00	1.2	0.07
HP-3	SW 1000	2.4	0.96	0.00	0.00	0.00	0.96	0.05
HP-4	W Settling Basin	2.6	0.88	0.00	0.00	0.00	0.88	0.05
HP-5	E 2506	5.1	1.8	0.02	0.00	0.00	1.8	0.10
HP-6	SW 3027	3.1	1.3	0.00	0.00	0.00	1.3	0.07
HP-7	W 7001	2.3	0.85	0.00	0.00	0.00	0.85	0.05
HP-8	Rock Quarry	2.7	0.83	0.00	0.00	0.00	0.83	0.04
HP-9	N Bethel Valley Rd.	2.4	0.75	0.00	0.00	0.00	0.75	0.04
HP-10	W 2075	2.9	1.7	0.00	0.00	0.00	1.7	0.07
HP-16	E 4500	2.8	1.1	0.00	0.00	0.00	1.1	0.06
HP-20	HFIR	2.8	0.92	0.00	0.00	0.00	0.92	0.05
Average		2.9	1.3	0.01	0.00	0.00	1.3	0.07
			Perimeter Area					
HP-31	Kerr Hollow Gate	1.5	1.6	0.00	0.00	0.00	1.6	0.03
HP-32	Midway Gate	2.2	1.9	0.00	0.00	0.00	1.9	0.03
HP-33	Gallaher Gate	1.3	1.6	0.00	0.00	0.00	1.6	0.03
HP-34	White Oak Dam	1.5	1.8	0.00	0.00	0.00	1.8	0.03
HP-35	Blair Gate	1.6	2.4	0.00	0.00	0.00	2.4	0.04
HP-36	Turnpike Gate	1.8	2.0	0.00	0.00	0.00	2.0	0.03
HP-37	Hickory Creek Bend	1.2	1.6	0.00	0.00	0.00	1.6	0.03
HP-38	E EGCR	1.7	1.6	0.00	0.00	0.00	1.6	0.04
HP-39	Townsite	1.5	1.0	0.00	0.00	0.00	1.0	0.02
Average		1.6	1.7	0.00	0.00	0.00	1.7	0.03
			Remote Area					
HP-51	Norris Dam	1.6	1.7	0.00	0.00	0.00	1.7	0.03
HP-52	Loudoun Dam	1.6	1.7	0.00	0.00	0.00	1.7	0.03
HP-53	Douglas Dam	1.5	1.2	0.00	0.00	0.00	1.2	0.02
HP-54	Cherokee Dam	1.4	2.0	0.00	0.00	0.00	2.0	0.03
HP-55	Watts Bar Dam	1.3	1.2	0.00	0.00	0.00	1.2	0.02
HP-56	Great Falls Dam	1.8	2.2	0.00	0.00	0.00	2.2	0.04
HP-57	Dale Hollow Dam	1.5	1.9	0.00	0.00	0.00	1.9	0.04
HP-58	Knoxville	1.7	2.1	0.00	0.00	0.00	2.1	0.04
Average		1.6	1.7	0.00	0.00	0.00	1.7	0.03

Table 5.2

Concentration of Radioactive Materials in Air
As Determined from Filter Paper Data - 1968
(System Average - by Weeks)

Week Number	Units of 10^{-13} µCi/cc			Week Number	Units of 10^{-13} µCi/cc		
	LAM's	PAM's	RAM's		LAM's	PAM's	RAM's
1	1.2	0.54	0.56	29	3.0	1.2	1.3
2	1.2	0.59	0.59	30	1.7	1.3	1.2
3	3.9	2.1	2.0	31	1.3	1.0	0.89
4	4.6	2.4	2.4	32	1.0	0.73	0.65
5	5.3	2.1	2.1	33	0.49	0.50	0.46
6	3.5	1.8	1.7	34	1.3	0.67	0.71
7	2.8	1.5	1.4	35	2.6	1.6	2.0
8	3.3	1.6	1.8	36	2.2	1.1	1.3
9	2.6	1.5	1.7	37	1.9	1.1	1.0
10	2.8	1.6	1.8	38	3.5	1.1	1.2
11	4.6	1.6	1.8	39	2.8	1.4	1.1
12	4.8	2.0	1.8	40	2.1	1.0	0.87
13	5.0	3.1	2.9	41	1.0	0.61	0.46
14	4.7	2.9	2.9	42	1.7	0.67	0.83
15	5.0	2.9	2.9	43	1.9	1.2	1.1
16	6.2	3.6	3.1	44	1.1	1.1	0.81
17	4.6	3.4	3.1	45	3.1	0.53	0.42
18	3.6	2.3	2.3	46	2.2	0.72	0.56
19	5.0	3.2	3.2	47	1.2	0.54	0.60
20	3.4	2.0	2.2	48	0.76	0.70	0.42
21	3.5	2.5	2.3	49	1.1	0.78	0.37
22	3.0	1.9	2.1	50	1.5	0.83	0.68
23	4.5	2.5	2.7	51	1.6	0.78	0.86
24	4.5	2.9	2.7	52	1.4	0.67	0.67
25	3.4	2.3	2.3				
26	2.9	2.1	1.9	Average			
27	4.2	2.2	2.4	1968	2.9	1.6	1.6
28	3.1	2.1	1.9	1967	2.2	1.1	1.0

Table 5.3

Radioparticulate Fallout Measurements
As Determined by Autoradiographic Techniques - 1968
(Gummed Paper Data - System Average by Weeks)

Week Number	Particles/ft^2			Week Number	Particles/ft^2		
	LAM's	PAM's	RAM's		LAM's	PAM's	RAM's
1	6.1	0.90	0.71	29	0.08	0.18	0.13
2	18.3	1.11	1.20	30	0.0	0.18	0.21
3	2.3	0.17	0.16	31	0.17	0.21	0.17
4	2.9	0.43	0.42	32	0.08	0.16	0.13
5	0.58	0.46	0.30	33	0.17	0.16	0.18
6	0.42	0.17	0.12	34	0.75	0.06	0.07
7	0.25	0.26	0.33	35	0.25	0.14	0.09
8	0.25	0.07	0.13	36	0.17	0.17	0.14
9	0.33	0.25	0.20	37	0.25	0.16	0.14
10	0.0	0.86	0.67	38	0.25	0.32	0.22
11	0.0	0.99	0.84	39	0.50	0.09	0.03
12	0.33	0.68	0.47	40	0.17	0.29	0.26
13	0.0	0.69	0.65	41	0.50	0.17	0.10
14	0.25	0.71	0.68	42	0.17	0.11	0.07
15	0.42	1.11	1.04	43	0.17	0.19	0.18
16	0.17	0.55	0.51	44	0.17	0.09	0.06
17	0.0	1.13	1.00	45	1.25	0.23	0.14
18	0.08	0.47	0.60	46	0.08	0.27	0.14
19	0.17	0.53	0.52	47	0.0	0.13	0.08
20	0.17	0.73	0.65	48	0.0	0.13	0.14
21	0.42	0.51	0.41	49	0.50	0.17	0.10
22	0.33	0.93	0.84	50	0.25	0.27	0.21
23	0.25	0.28	0.15	51	0.17	0.31	0.09
24	0.25	0.49	0.32	52	0.0	0.11	0.12
25	0.08	0.17	0.09				
26	0.17	0.26	0.20	Average			
27	0.17	0.31	0.26	1968	0.79	0.57	0.38
28	0.0	0.44	0.38	1967	12.8	12.3	6.3

Table 5.4 Radioparticulate Fallout--1968
(Gummed Paper Data--Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-4} $\mu Ci/ft^2$	No. of Particles by Activity Ranges				Total Particles Per Sq. Ft.
			$< 10^5$ d/24 hr	10^5–10^6 d/24 hr	10^6–10^7 d/24 hr	$> 10^7$ d/24 hr	
			Laboratory Area				
HP-1	S 3587	0.40	0.67	0.00	0.00	0.00	0.67
HP-2	NE 3025	0.74	0.83	0.02	0.00	0.00	0.85
HP-3	SW 1000	0.47	0.67	0.02	0.00	0.00	0.69
HP-4	W Settling Basin	0.58	0.56	0.00	0.00	0.00	0.56
HP-5	E 2506	0.70	0.79	0.00	0.00	0.00	0.81
HP-6	SW 3027	0.62	1.17	0.02	0.00	0.00	1.19
HP-7	W 7001	0.36	0.65	0.00	0.00	0.00	0.65
HP-8	Rock Quarry	0.41	0.48	0.00	0.00	0.00	0.48
HP-9	N Bethel Valley Rd.	0.34	0.56	0.00	0.00	0.00	0.56
HP-10	W 2075	0.78	1.17	0.02	0.00	0.00	1.19
*HP-16	E 4500	0.50	1.10	0.00	0.00	0.00	1.10
*HP-20	HFIR	0.44	0.65	0.00	0.00	0.00	0.65
Average		0.53	0.78	0.01	0.00	0.00	0.79
			Perimeter Area				
HP-31	Kerr Hollow Gate	0.39	0.58	0.00	0.00	0.00	0.58
HP-32	Midway Gate	0.55	1.37	0.00	0.00	0.00	1.37
HP-33	Gallaher Gate	0.33	0.40	0.00	0.00	0.00	0.40
HP-34	White Oak Dam	0.37	0.55	0.00	0.00	0.00	0.55
HP-35	Blair Gate	0.33	0.54	0.00	0.00	0.00	0.54
HP-36	Turnpike Gate	0.41	0.48	0.00	0.00	0.00	0.48
HP-37	Hickory Creek Bend	0.32	0.50	0.00	0.00	0.00	0.50
HP-38	E EGCR	0.31	0.26	0.00	0.00	0.00	0.26
HP-39	Townsite	0.44	0.44	0.00	0.00	0.00	0.44
Average		0.38	0.57	0.00	0.00	0.00	0.57
			Remote Area				
HP-51	Norris Dam	0.34	0.35	0.00	0.00	0.00	0.35
HP-52	Loudoun Dam	0.31	0.31	0.00	0.00	0.00	0.31
HP-53	Douglas Dam	0.29	0.31	0.00	0.00	0.00	0.31
HP-54	Cherokee Dam	0.32	0.63	0.00	0.00	0.00	0.63
HP-55	Watts Bar Dam	0.32	0.12	0.00	0.00	0.00	0.12
HP-56	Great Falls Dam	0.31	0.73	0.02	0.00	0.00	0.75
HP-57	Dale Hollow Dam	0.36	0.31	0.00	0.00	0.00	0.31
HP-58	Knoxville	0.38	0.25	0.00	0.00	0.00	0.25
Average		0.33	0.38	0.00	0.00	0.00	0.38

*Installed July, 1966.

Table 5.5

Discharge of ^{131}I from ORNL Stacks
and Weekly Average Concentrations of ^{131}I in Air and Milk - 1968

Week No.	PAM's Air Concentration Units of 10^{-12} μCi/cc	Milk Concentration pCi/l	Stack Discharge millicuries[a]	Week No.	PAM's Air Concentration Units of 10^{-12} μCi/cc	Milk Concentration pCi/l	Stack Discharge millicuries[a]
1	0.017	8.6	330	29	0.020	5.0	320
2	0.011	6.9	205	30	0.013	5.0	630
3	0.014	5.0	156	31	0.070	5.0	490
4	0.008	11.9	92	32	0.016	5.0	487
5	0.013	11.2	70	33	0.008	5.0	254
6	0.011	18.6	271	34	0.009	5.0	68
7	0.010	5.0	229	35	0.006	5.0	64
8	0.006	5.0	72	36	0.005	5.0	48
9	0.019	5.0	437	37	0.006	5.0	35
10	0.014	5.0	485	38	0.005	5.0	37
11	0.007	5.0	133	39	0.007	5.0	120
12	0.006	5.0	168	40	0.004	5.0	41
13	0.013	5.0	118	41	0.013	6.4	50
14	0.025	5.0	329	42	0.043	5.0	49
15	0.010	5.0	400	43	0.023	5.0	21
16	0.015	5.0	158	44	0.006	5.0	45
17	0.009	5.0	225	45	0.014	5.0	50
18	0.009	5.0	296	46	0.023	5.0	206
19	0.010	5.0	331	47	0.027	5.0	74
20	0.009	5.0	374	48	0.010	5.0	113
21	0.007	5.0	1269	49	0.006	5.0	13
22	0.010	5.0	198	50	0.010	5.0	32
23	0.004	5.0	177	51	0.007	5.0	33
24	0.015	5.0	356	52	0.006	5.0	15
25	0.018	5.0	113				
26	0.006	8.1	41	Total			10,377
27	0.006	5.0	22				
28	0.025	5.0	27	Average	0.013	5.7	

[a]Data furnished by Laboratory Facilities Department.

Table 5.6 Concentration of ^{131}I in Air—1968

Location	Units of 10^{-12} μCi/cc		
	Maximum	Minimum[a]	Average
ORNL Plant Area	2.6	< 0.010	0.133
Perimeter Area	0.29	< 0.010	0.013

[a]Minimum detectable amount of ^{131}I is 20 d/m. At the average sampling rate this corresponds to approximately 0.010×10^{-12} μCi/cc on the perimeter monitors and approximately 0.020×10^{-12} μCi/cc on the Plant monitors. In averaging, one-half of this value, 10 d/m is used for all samples showing a total amount of ^{131}I less than 20 d/m.

Table 5.7 Concentration of Radioactive Materials in Rainwater—1968
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rainwater, μCi/ml
	Laboratory Area	
HP-7	West 7001	0.27×10^{-7} μCi/ml
	Perimeter Area	
HP-31	Kerr Hollow Gate	0.27×10^{-7} μCi/ml
HP-32	Midway Gate	0.32
HP-33	Gallaher Gate	0.33
HP-34	White Oak Dam	0.36
HP-35	Blair Gate	0.35
HP-36	Turnpike Gate	0.33
HP-37	Hickory Creek Bend	0.40
HP-38	E EGCR	0.41
HP-39	Townsite	0.28
Average		0.34×10^{-7} μCi/ml
	Remote Area	
HP-51	Norris Dam	0.43×10^{-7} μCi/ml
HP-52	Loudoun Dam	0.34
HP-53	Douglas Dam	0.32
HP-54	Cherokee Dam	0.42
HP-55	Watts Bar Dam	0.37
HP-56	Great Falls Dam	0.31
HP-57	Dale Hollow Dam	0.22
HP-58	Knoxville	0.24
Average		0.33×10^{-7} μCi/ml

Table 5.8

Weekly Average Concentration of Radioactivity in Rainwater - 1968

Units of 10^{-7} μCi/ml

Week Number	LAM's	PAM's	RAM's	Week Number	LAM's	PAM's	RAM's
1	0.25	0.21	0.25	29	0.02	0.20	0.27
2	0.25	0.23	0.33	30	0.11	0.22	0.30
3	No Rain	No Rain	No Rain	31	0.02	0.21	0.45
4	0.40	0.55	1.02	32	0.14	0.17	0.23
5	0.33	0.50	0.52	33	No Rain	No Rain	0.08
6	No Rain	No Rain	No Rain	34	No Rain	No Rain	0.35
7	No Rain	No Rain	No Rain	35	No Rain	0.24	0.43
8	No Rain	No Rain	No Rain	36	0.07	0.18	0.11
9	0.17	0.32	0.19	37	0.18	0.25	0.38
10	0.54	0.79	0.88	38	0.05	0.15	0.13
11	0.32	0.45	0.32	39	No Rain	No Rain	No Rain
12	0.14	0.31	0.23	40	0.25	0.10	0.14
13	0.52	0.60	0.65	41	No Rain	0.16	0.23
14	0.47	0.39	0.37	42	0	0.12	0.10
15	0.52	0.58	0.63	43	No Rain	No Rain	No Rain
16	No Rain	No Rain	No Rain	44	0.37	0.32	0.31
17	0.27	0.47	0.61	45	0.18	0.20	0.13
18	0.11	0.35	0.34	46	0.06	0.20	0.18
19	0.96	0.49	0.89	47	0.20	0.25	0.42
20	0.11	0.22	0.09	48	0.11	0.15	0.14
21	0.11	0.17	0.15	49	0.28	0.17	0.14
22	0.13	0.47	0.46	50	0.09	0.19	0.30
23	0.72	0.40	0.40	51	0.26	0.39	0.19
24	0.56	0.70	0.73	52	0.14	0.15	0.19
25	No Rain	0.30	0.49				
26	0.14	0.16	0.17	Average			
27	0.18	0.46	0.68	1968	0.27	0.34	0.33
28	0.16	0.31	0.37	1967	0.26	0.20	0.19

Table 5.9 Liquid Waste Discharged from White Oak Creek—1968

	Curies	
	Total for Year	Weekly Average
Beta Activity	16	0.31
Transuranic Alpha Emitters	0.04	0.008

Table 5.10 Yearly Discharges of Radionuclides to Clinch River (Curies)

Year	Gross Beta	^{137}Cs	^{106}Ru	^{90}Sr	TRE*(-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co
1949	718	77	110	150	77	18	180	22	77	
1950	191	19	23	38	30		15	42	19	
1951	101	20	18	29	11		4.5	2.2	18	
1952	214	9.9	15	72	26	23	19	18	20	
1953	304	6.4	26	130	110	6.7	7.6	3.6	2.1	
1954	384	22	11	140	160	24	14	9.2	3.5	
1955	437	63	31	93	150	85	5.2	5.7	7.0	6.6
1956	582	170	29	100	140	59	12	15	3.5	46
1957	397	89	60	83	110	13	23	7.1	1.2	4.8
1958	544	55	42	150	240	30	6.0	6.0	8.2	8.7
1959	937	76	520	60	94	48	27	30	0.5	77
1960	2190	31	1900	28	48	27	38	45	5.3	72
1961	2230	15	2000	22	24	4.2	20	70	3.7	31
1962	1440	5.6	1400	9.4	11	1.2	2.2	7.7	0.36	14
1963	470	3.6	430	7.8	9.4	1.5	0.34	0.71	0.44	14
1964	234	6.0	191	6.6	13	0.3	0.16	0.07	0.29	15
1965	95	2.1	69	3.4	5.9	0.1	0.33	0.33	0.20	12
1966	48	1.6	29	3.0	4.9	0.1	0.67	0.67	0.24	7
1967	40	2.7	17	5.1	8.5	0.2	0.49	0.49	0.91	3
1968	16	1.1	5	2.8	4.4	0.03	0.27	0.27	0.31	1

*Tri-Valent Rare Earths.

Table 5.11 Radioactivity in Clinch River—1968

Location	Concentration of Radionuclides of Primary Concern in Units of 10^{-8} μCi/ml						Average Concentration of Total Radioactivity	$(MPC)_w$ [a]	% of $(MPC)_w$
	^{90}Sr	^{144}Ce	^{137}Cs	$^{103-106}Ru$	^{60}Co	^{95}Zr-^{95}Nb	10^{-8} μCi/ml	10^{-6} μCi/ml	
CRM 23.1[b]	0.04	0.02	*	0.06	0.44	*	0.57	3.4	0.17
CRM 20.8[c]	0.06	< 0.01	0.02	0.11	0.02	< 0.01	1.0	1.2	0.83
CRM 4.5[b]	0.15	0.04	0.14	0.07	0.27	0.02	0.69	1.3	0.52

[a] Weighted average $(MPC)_w$ calculated for the mixture, using $(MPC)_w$ values for specific radionuclides specified by AEC Manual, Chapter 0524, Appendix, Annex 1, Table II.

[b] Measured values.

[c] Values given for this location are calculated values based on the levels of waste released and the dilution afforded by the river; they do not include amounts of radioactive material (e.g., fallout) that may enter the river upstream from CRM 20.8.

* None detected.

Table 5.12

Estimated Percent $(MPC)_w$ of Radioactivity in Clinch River Water Below the Mouth of White Oak Creek - 1968

Week Number	% MPC_w	Week Number	% MPC_w
1	0.21	29	0.15
2	0.30	30	0.10
3	0.30	31	0.10
4	0.12	32	0.15
5	0.26	33	0.14
6	0.25	34	0.12
7	0.27	35	0.30
8	0.23	36	0.30
9	0.25	37	0.30
10	0.42	38	0.30
11	1.51	39	0.30
12	2.11	40	0.63
13	1.95	41	0.31
14	2.47	42	0.24
15	4.49	43	0.12
16	2.35	44	0.12
17	1.03	45	0.06
18	1.35	46	0.10
19	3.18	47	0.33
20	1.10	48	0.33
21	0.58	49	0.13
22	3.27	50	0.17
23	1.47	51	0.62
24	0.84	52	1.65
25	0.14		
26	0.38	Average	
27	0.34	1968	0.83
28	0.17	1967	0.71

Table 5.13

Weekly Average Concentration of ^{90}Sr in Raw Milk
In the Immediate Environs of Oak Ridge - 1968

Week Number	pCi/l	Week Number	pCi/l
1	20.2	29	18.9
2	22.3	30	21.6
3	16.8	31	20.8
4	20.9	32	18.6
5	23.1	33	21.1
6	20.9	34	19.2
7	19.7	35	16.2
8	23.3	36	19.9
9	18.9	37	25.2
10	20.8	38	16.3
11	17.0	39	18.4
12	21.4	40	18.5
13	22.2	41	16.5
14	23.1	42	18.0
15	21.3	43	19.5
16	22.6	44	18.7
17	22.4	45	17.9
18	21.9	46	15.7
19	21.8	47	19.4
20	23.3	48	19.9
21	21.1	49	18.0
22	20.9	50	16.7
23	23.0	51	21.6
24	21.1	52	18.4
25	20.6		
26	19.3		
27	19.3		
28	18.6	Average	20.0

Table 5.14 Radionuclide Content of Clinch River Fish

Species	Year	pCi/kg Fresh Weight		
		^{90}Sr	^{106}Ru	^{137}Cs
White Crappie	1965	14	284	199
	1966	9.4	381	87
	1967	27	*	387
	1968	7.3	*	331
Smallmouth Buffalo	1965	32	6467	194
	1966	94	185	1303
	1967	27	122	402
	1968	16	*	132
Gizzard Shad	1966	2028	513	1453
	1967	118	*	399
	1968	473	*	559

*None detected.

Table 5.15 Estimated Percentage of MPI That Man May Attain by Eating Clinch River Fish

Year	^{90}Sr	^{106}Ru	^{137}Cs	Total (%)
		Smallmouth Buffalo		
1965	0.24	1.4	0.020	1.66
1966	0.66	0.039	0.137	0.836
1967	0.19	0.026	0.044	0.260
1968	0.11	*	0.014	0.126
		White Crappie		
1965	0.099	0.060	0.021	0.180
1966	0.066	0.082	0.009	0.157
1967	0.19	*	0.041	0.231
1968	0.051	*	0.035	0.086

*None detected.

ORNL-DWG. 66-2218

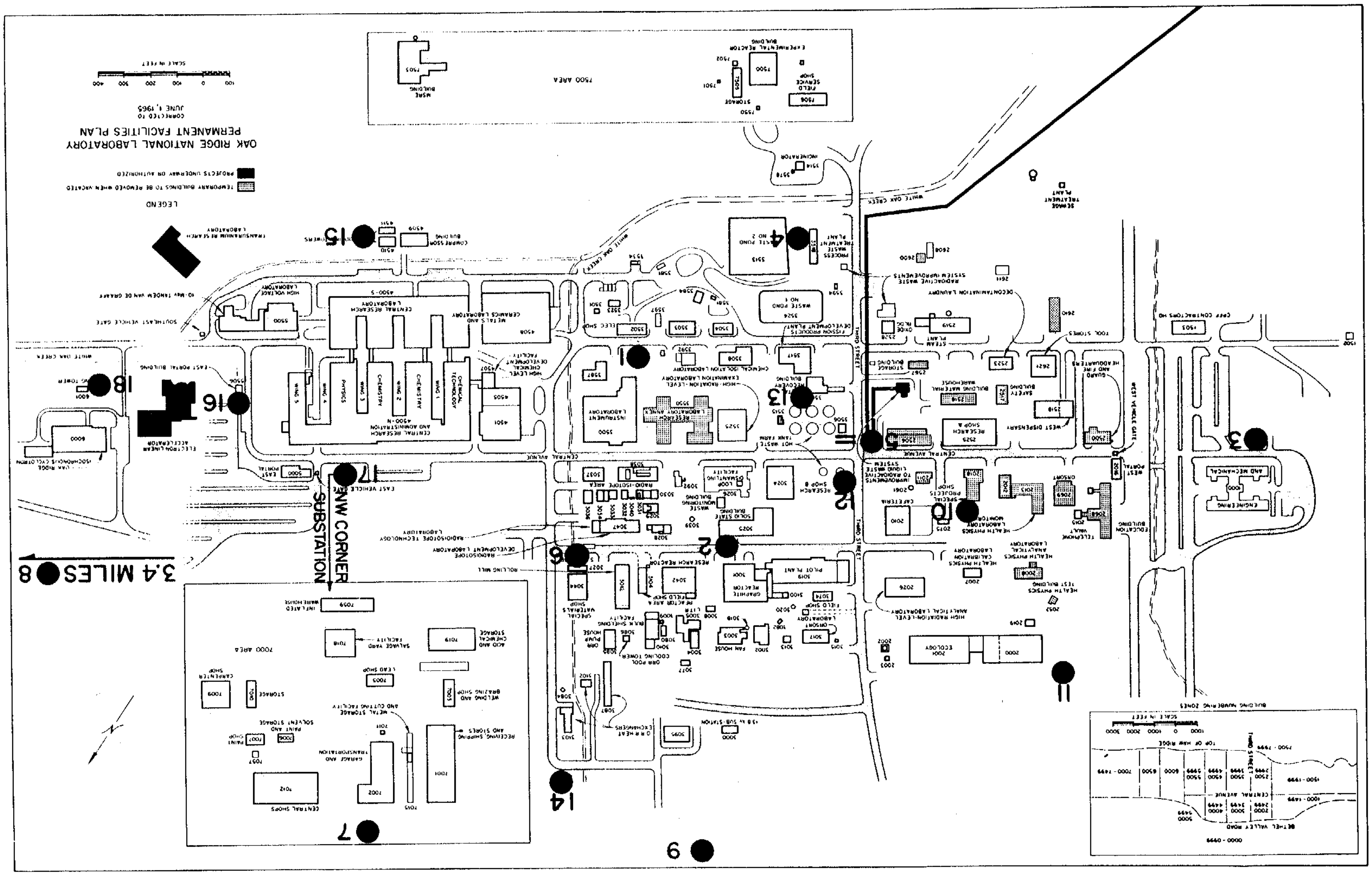

Fig. 5.1 Map of X-10 Area Showing the Approximate Location of 18 of 22 of the Local Monitoring Stations Constituting the LAM Network.

ORNL-DWG. 66-1718

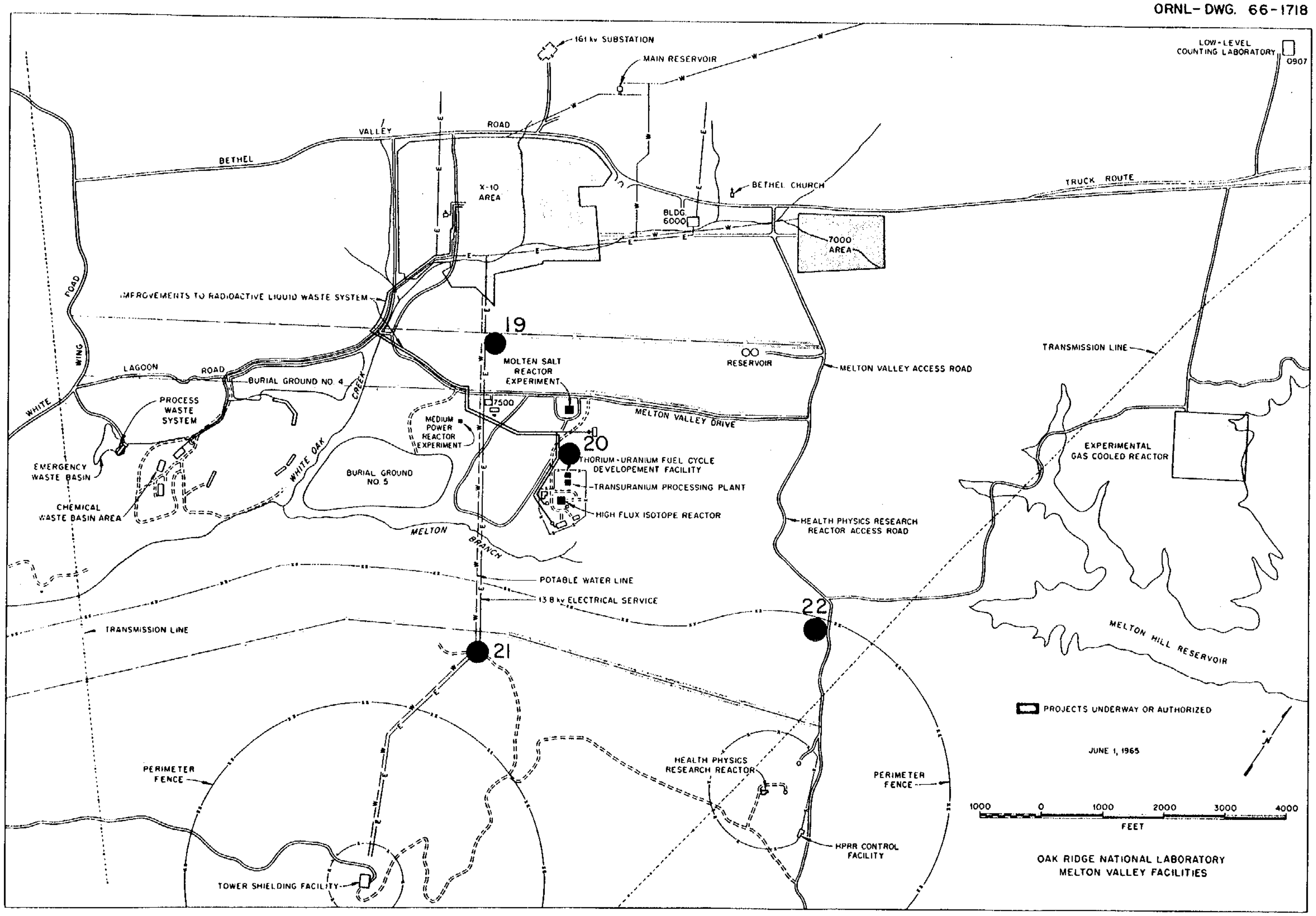

Fig. 5.2 Map of Laboratory Area Showing the Approximate Location of 4 of the 22 Local Monitoring Stations Constituting the LAM Network.

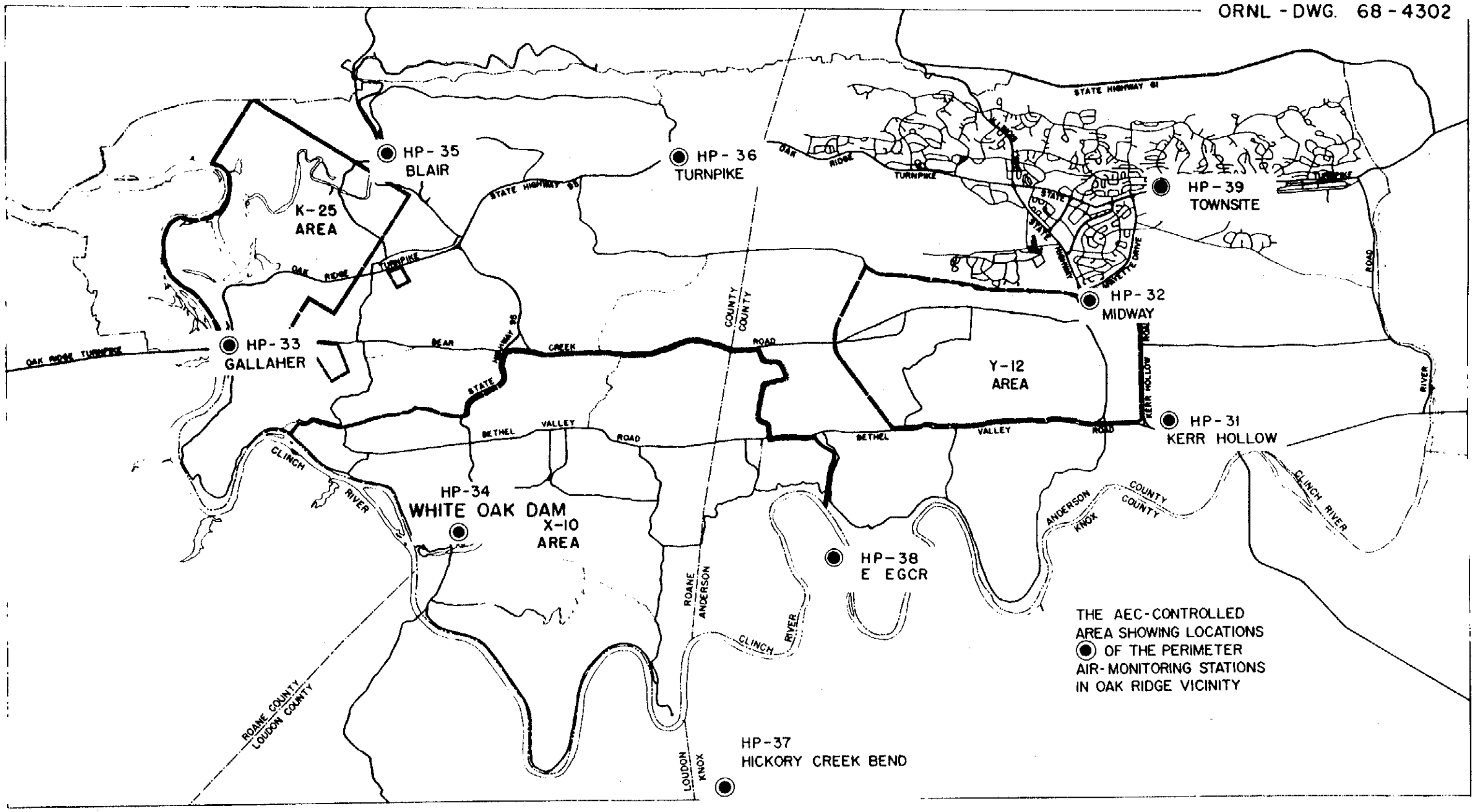

Fig. 5.3 Map of the AEC Controlled Area and Vicinity Showing the Approximate Location of the Perimeter Air Monitoring Stations Constituting the PAM Network.

ORNL-DWG 66-1719

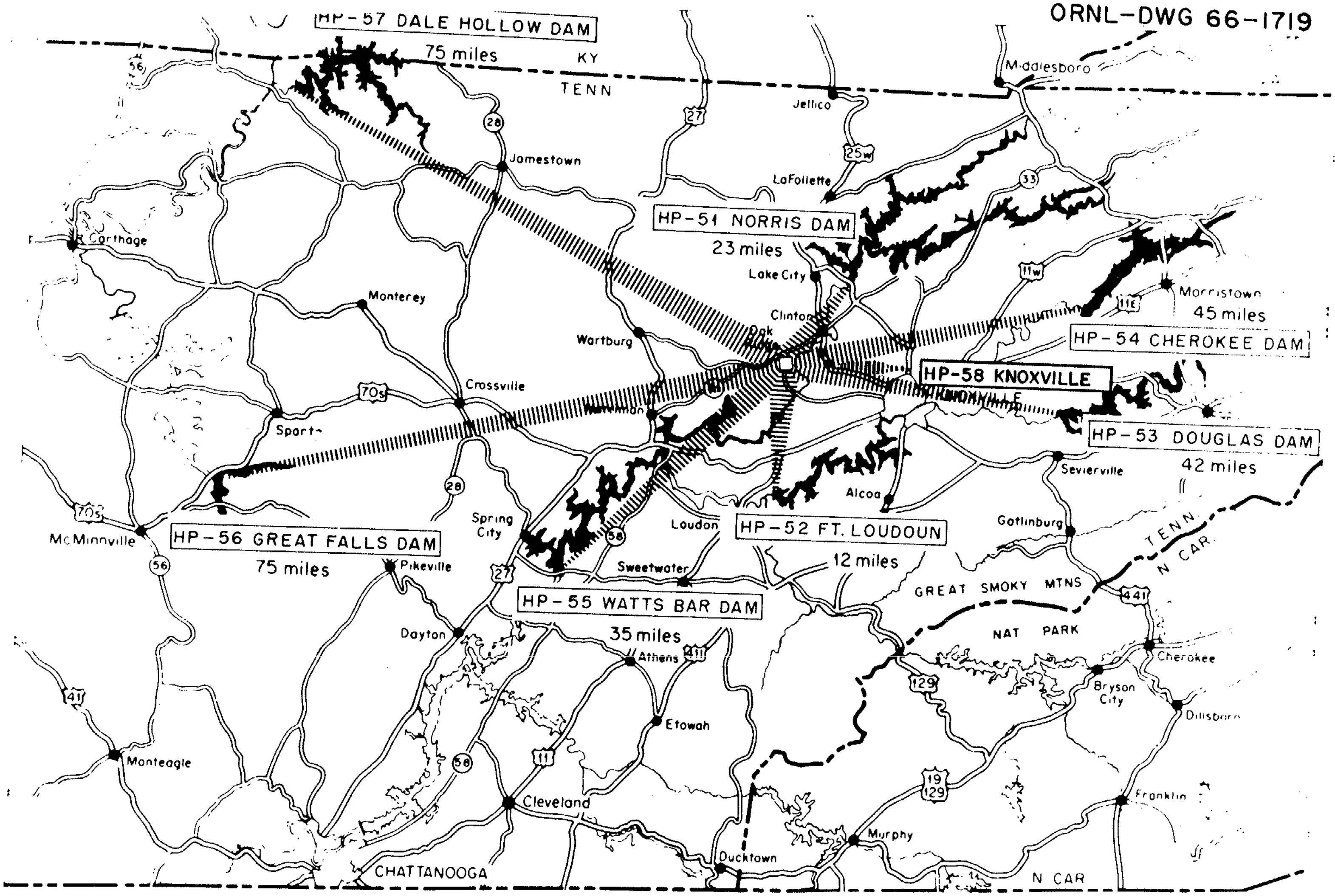

Fig. 5.4 Map of a Section of the East Tennessee Area Showing TVA and U. S. Corps of Engineers Dam Sites at which are Located the Remote Air Monitoring Stations Constituting the RAM Network.

ORNL-DWG. 66-2216R

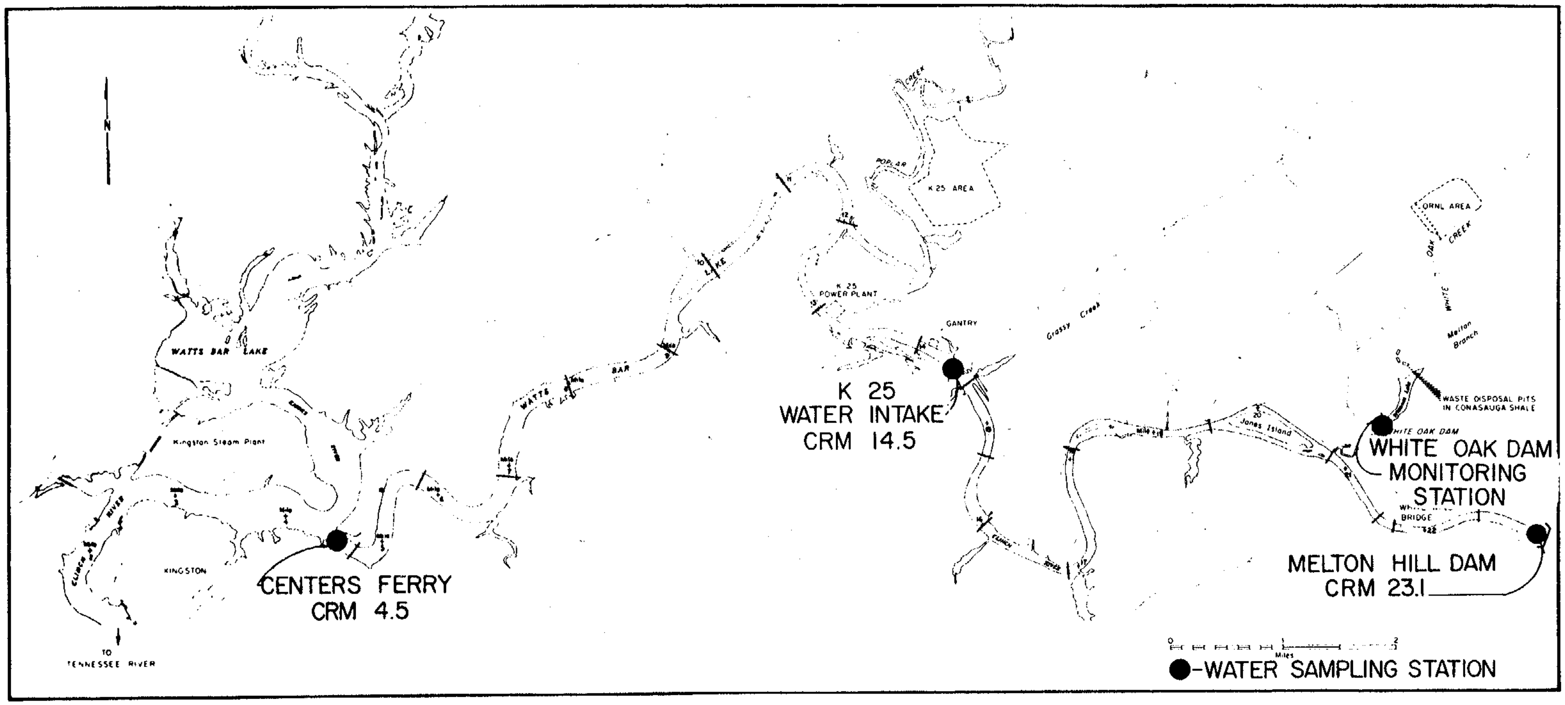

Fig. 5.5 Map Showing Water Sampling Locations in the East Tennessee Area.

ORNL—DWG. 64-3713 R1

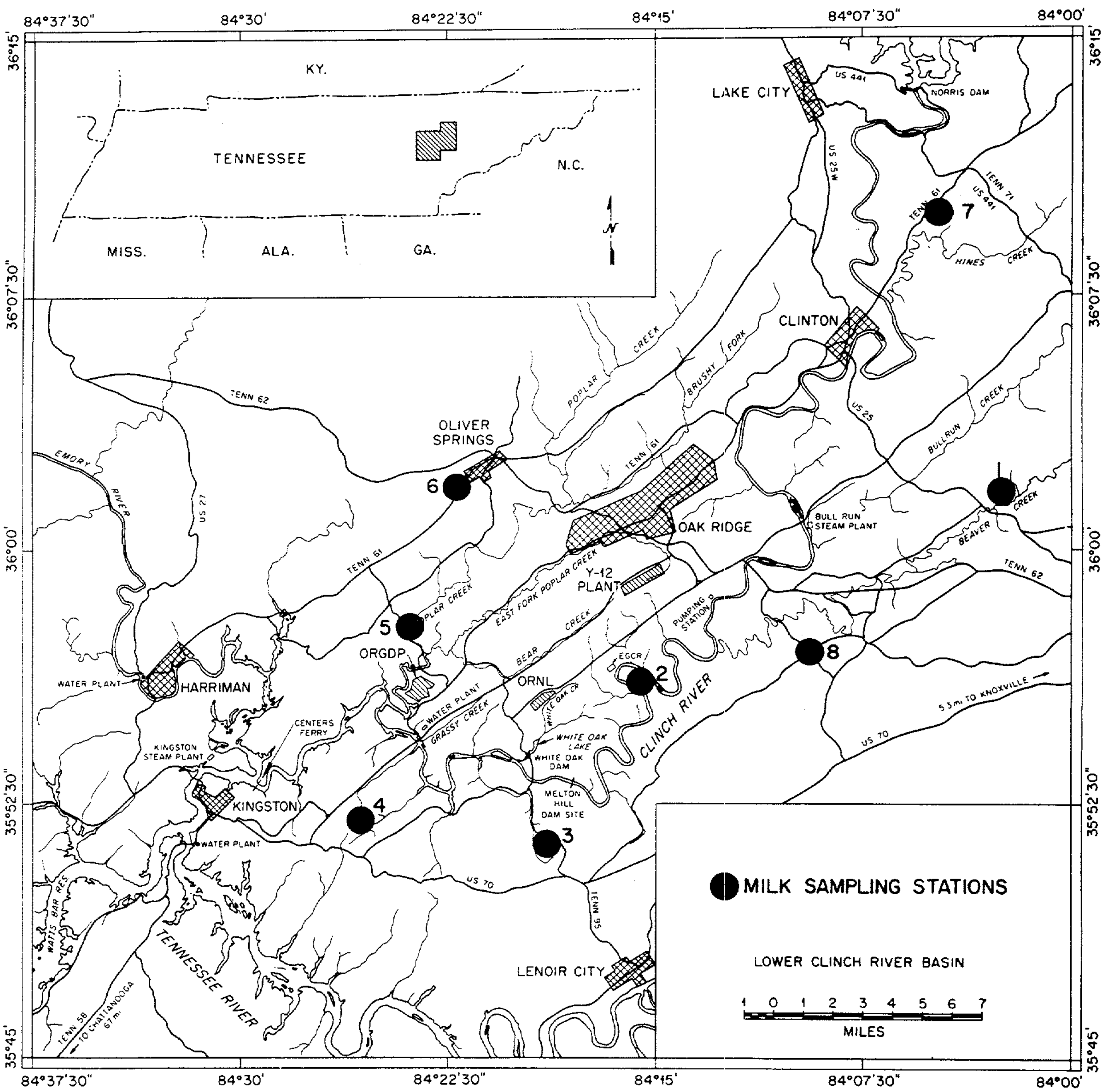

Fig. 5.6 Map Showing Milk Sampling Stations in the East Tennessee Area.

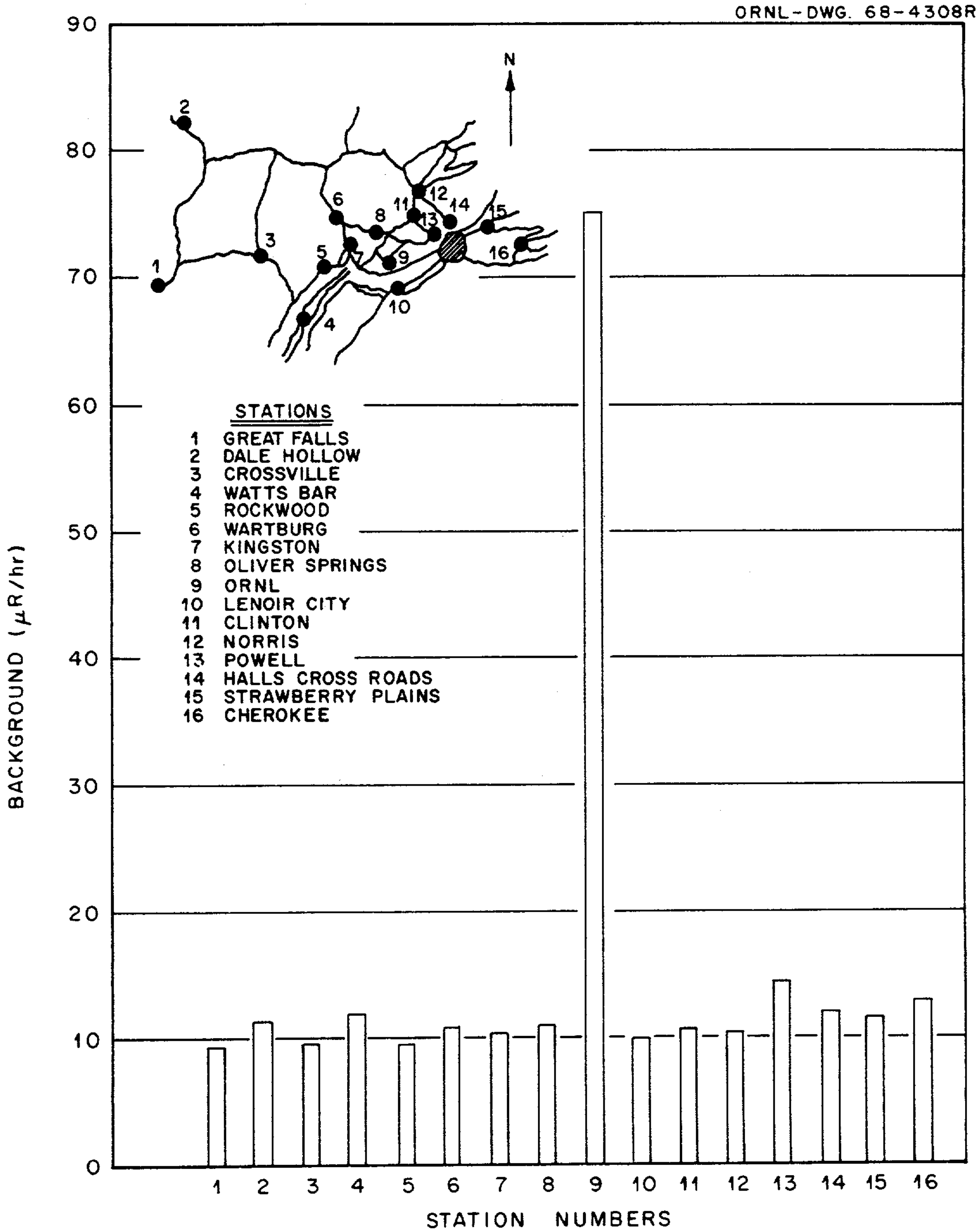

Fig. 5.7 Radiation Measurements Taken During 1968, 3 ft. Above the Ground Surface out to Distances of 75 Miles from ORNL.

ORNL-DWG. 67-3514R2

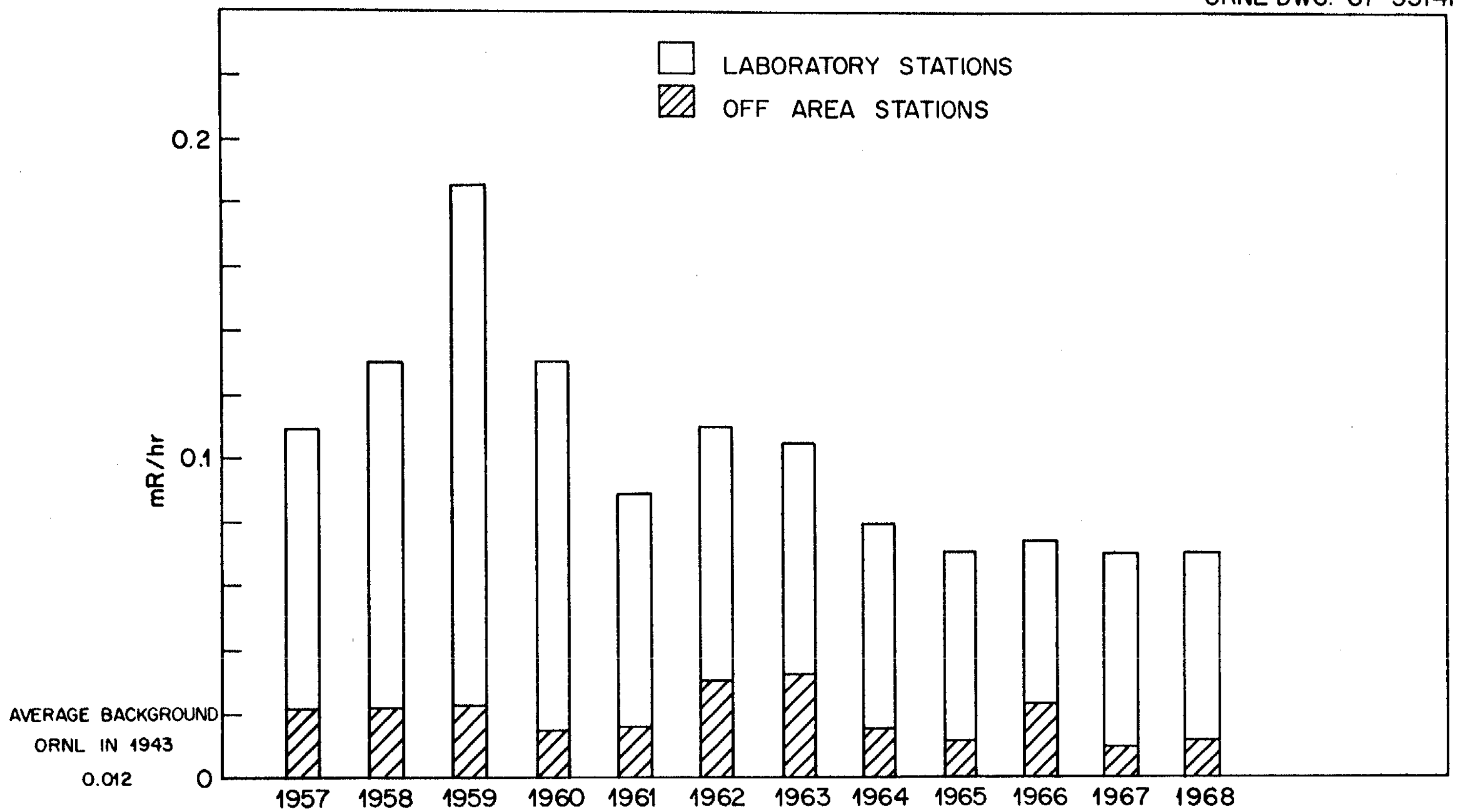

Fig. 5.8 Radiation Measurements Taken 3 ft. Above the Ground Surfaces at ORNL Compared with Like Measurements Taken Elsewhere within the AEC Controlled Area for the Years 1957–1968.

6.0 PERSONNEL MONITORING

It is the policy of the Oak Ridge National Laboratory to monitor the radiation exposure of all persons who enter Laboratory areas where there is a likelihood of radiation exposure. Dose analysis is accomplished mainly through the use of personnel meters, bio-assays, and in vivo counting (whole body counter) techniques.

6.1 Dose Analysis Summary, 1968

6.1.1. External Exposures - No employee received a whole body radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest whole body dose received by an employee was about 4.7 rem or 39 percent of the maximum permissible annual dose. The range of doses for persons using ORNL badge-meters is shown in Table 6.1.

As of December 31, 1968, no employee had a cumulative whole body dose which exceeded the recommended maximum permissible dose as based on the age proration formula 5(N - 18) (Table 6.2). Only one employee had an average annual exposure rate that exceeded 5 rem per year of employment (Table 6.3).

The highest cumulative dose to the skin of the whole body received by an employee during 1968 was about 13 rem or 43 percent of the maximum permissible annual skin dose of 30 rem.

As of December 31, 1968, the highest cumulative dose of whole body radiation received by an employee was approximately 91 rem. This dose was accrued over an employment period of about 25 years and represented an average exposure of about 3.7 rem per year.

The highest cumulative hand exposure recorded during 1968 was about 12 rem or 16 percent of the recommended maximum permissible annual dose to the extremities.

The average of the ten highest whole body doses of ORNL employees for each of the years 1962 through 1968 are shown in Figure 6.1. The highest individual dose for each of those years is shown also.

The average annual dose to ORNL employees for the years 1962 through 1968 is the subject of Figure 6.2. This rather arbitrary quantity is obtained by dividing the sum of all doses for the year by the number of employees involved.

6.1.2 Internal Exposures - During 1968 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half a maximum permissible body burden.[5]

[5] NBS Handbook 69 values are the basis for these determinations.

Three employees continued to have estimated body burdens of transuranic alpha emitters (mainly ^{239}Pu) of 35 to 45 percent of the recommended maximum permissible value.[6] The ICRP recommends, Publication 6, paragraph 86(a), individuals who exceed 50 percent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

6.2 External Dose Techniques

6.2.1 Film Meters - Film meters are issued to all persons who have access to ORNL facilities in which there is a likelihood of radiation exposures for which monitoring is required. Either an ORNL badge-meter (Figure 6.3) or a temporary pass-meter (Figure 6.4) may be used. Badge-meters are assigned to all ORNL employees, and to certain other persons who are authorized to enter ORNL facilities. Temporary pass-meters may be issued in lieu of badge-meters for short-term use.

NTA (nuclear track) film packets are included in all film meters. The NTA films are processed routinely if the badge-meter is assigned to an individual who normally works where there may be exposure to neutrons; otherwise the films would be processed only in the event of a nuclear accident.

Beta-gamma sensitive films from badge-meters issued to full-time employees are processed routinely each calendar quarter (or more frequently if necessary). Films used in other meters are processed as conditions of use may require. Films from meters issued to visitors are processed if there is a likelihood that a radiation exposure was incurred.

High-level radiation dosimetry components of the badge-meters (sulfur, gold, indium, and metaphosphate glass) are for use in the event that doses exceed the capability of the monitoring films.

For each ORNL division which had one or more employees who sustained a dose greater than 1 rem for the year, the number of employees so exposed are displayed in Figure 6.5. It may be noted that only ten (of 29) divisions had employees with doses greater than 1 rem, only seven had employees with doses greater than 2 rem, and only four had employees with doses greater than 3 rem.

6.2.2 Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket

[6] AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 percent of a maximum permissible limit.

meter pairs are processed each day by Health Physics technicians and readings of 20 mR or more are reported daily to supervision. Pocket meters are used for a day-to-day record of integrated exposure.

6.2.3 Hand Exposure Meters - Hand exposure meters (Figure 6.6) are film-loaded finger rings used to measure hand exposure. Hand exposure meters are issued to persons for use during operations where it is likely that the hand dose is such as to exceed 1 rem during the week. They are issued and collected by Radiation and Safety Surveys Section personnel who determine the need for this type of monitoring and arrange for a processing schedule.

6.2.4 Metering Resume - Shown in Table 6.4 are the quantities of personnel metering devices used and processed during 1968. The number of films processed is less than the number issued, because those which are issued for accident dosimetry only are not processed unless there was a likelihood of exposure.

6.3 Internal Dose Techniques

6.3.1 Bio-Assays - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed is based upon each specific radioisotope and the exposure potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible, and only quantitative analyses for predetermined isotopes are performed routinely.

In most cases bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu. An estimate of dose is made for all cases in which it appears that one-third of a body burden, averaged over a calendar year, may be exceeded.

6.3.2 Whole Body Counter - The whole body counter (an in vivo gamma spectrometer) may be used for determing internally deposited quantities of most of the gamma ray-emitting substances, and some of the more energetic beta-emitting substances. Thus, it provides a direct method of determining body burdens of those substances.

6.4 Records and Reports

Most records and reports are prepared by automatic data processing (ADP) techniques through the use of high-speed digital computer systems. The IBM 7090, located at the Central Data Processing Facility (CDPF), provides routine weekly, quarterly, and annual reports involving external dose data. A typical weekly report is shown in Figure 6.7; a typical quarterly report is shown in Figure 6.8. A CDC 1604, operated by the ORNL Math Panel, is used to prepare the weekly pocket meter report (Figure 6.9) as well as the weekly, quarterly, and annual bio-assay reports. A sample of the Weekly Bio-Assay Sample Status Report is shown in Figure 6.10.

A quarterly and an annual report (Figure 6.11) based on results of analysis by the whole body counter (IVGS) are prepared by the IBM 360 at CDPF.

A quarterly and an annual report of occupational injuries is processed by the IBM 360 at ORNL.

An individual external dose summary (Figure 6.12) is prepared annually by updating on the IBM 360 at CDPF.

Body burden estimates of ^{239}Pu are prepared in report form (usually quarterly) by use of the IBM 7090 at CDPF.

Permanent files are maintained at Health Physics and Safety Headquarters for each individual who is assigned an ORNL photo-badge-meter. An IBM card cross-indexing system is maintained at the principal monitoring stations for the purpose of expediting meter assignments. These IBM cards are compatible with the various computer programs and provide for the internal audit of all personnel monitoring record data.

Copies of the ADP reports, both temporary and final, are maintained for both the internal and external dose programs. Data used in the ADP program are stored on computer quality magnetic tapes. Data pertinent to the work of the dosimetry groups and information used in the non-ADP reports are maintained in record form by the Dose Data Group.

Table 6.1 Dose Data Summary for Laboratory Population Involving Exposure to Whole Body Radiation - 1968

Group	Number of Rem Doses in Each Range							Total
	0-1	1-2	2-3	3-4	4-5	5-6	6 up	
ORNL Employees	5701	103	39	9	5	0	0	5857
ORNL-Badged Non-Employees	905	0	0	0	0	0	0	905
TOTAL	6606	103	39	9	5	0	0	6762

Table 6.2 Average Rem Per Year Since Age 18 - 1968

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5850	7	0	0	5857

Table 6.3 Average Rem Per Year of Employment at ORNL - 1968

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5827	29	1	0	5857

Table 6.4 Personnel Meter Services

	1966	1967	1968
A. Pocket Meter Usage			
1. Number of Pairs Used			
ORNL	156,676	150,748	143,572
CPFF	17,108	19,344	5,564
Total	173,784	170,092	149,136
2. Average Number of Users per Quarter			
ORNL	1,372	1,408	1,273
CPFF	213	252	94
Total	1,585	1,660	1,367
B. Film Usage			
1. Films Used in Photo-Badge-Meters			
Beta-Gamma	21,760	20,800	22,100
NTA	10,670	10,300	10,940
2. Films Used in Temporary Meters			
Beta-Gamma	7,790	4,930	8,850
NTA	2,520	1,600	2,860
C. Films Processed for Monitoring Data			
1. Beta-Gamma	22,190	21,150	22,720
2. NTA	2,470	1,580	1,190
3. Hand Meter	1,940	2,490	1,110

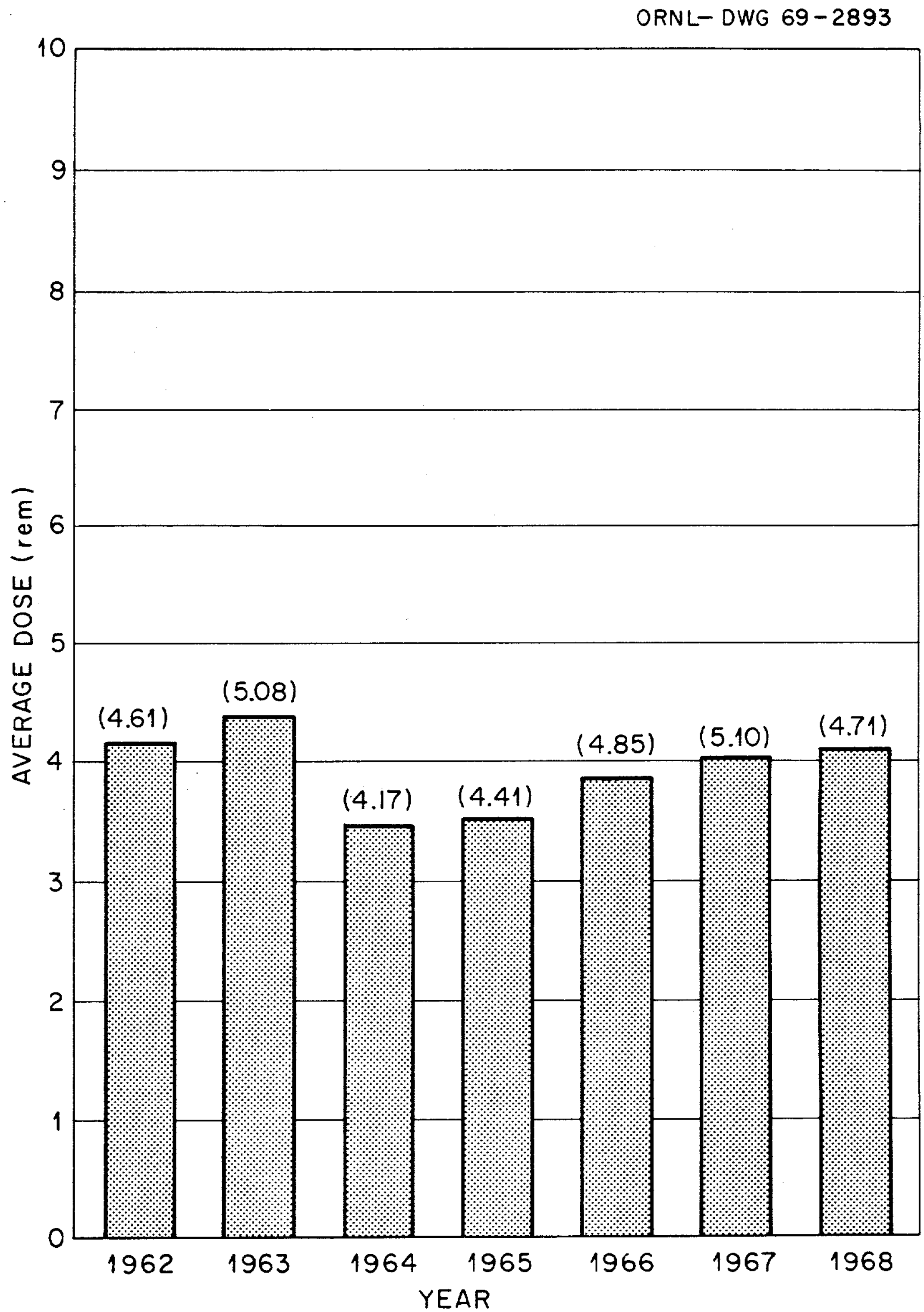

Fig. 6.1 Average of the Ten Highest Annual Whole Body Doses by Year (The Highest Individual Dose Shown in Parentheses).

ORNL—DWG 69-2894

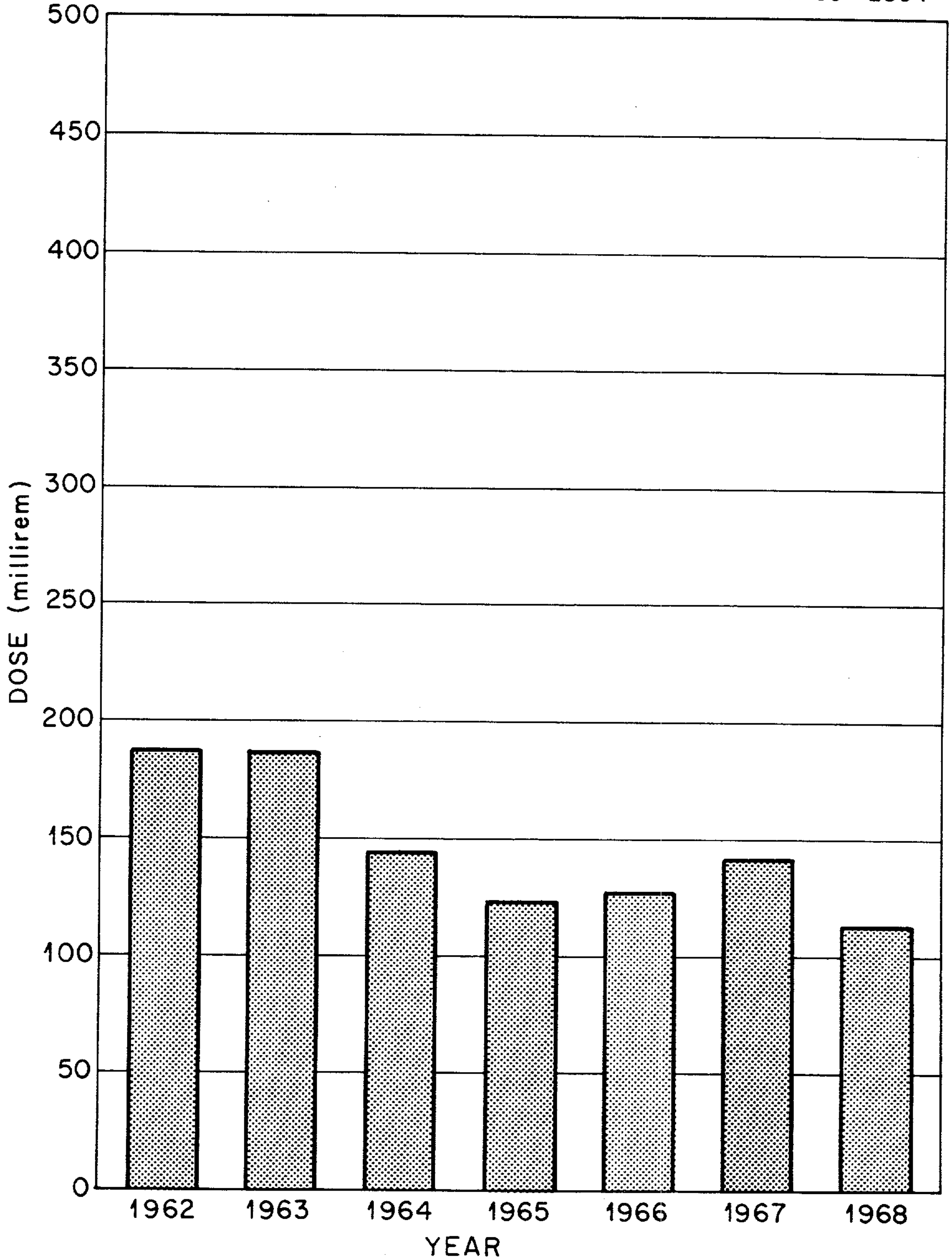

Fig. 6.2 Average Annual Whole Body Dose to the Average ORNL Employee.

ORNL–LR–DWG 50112R2

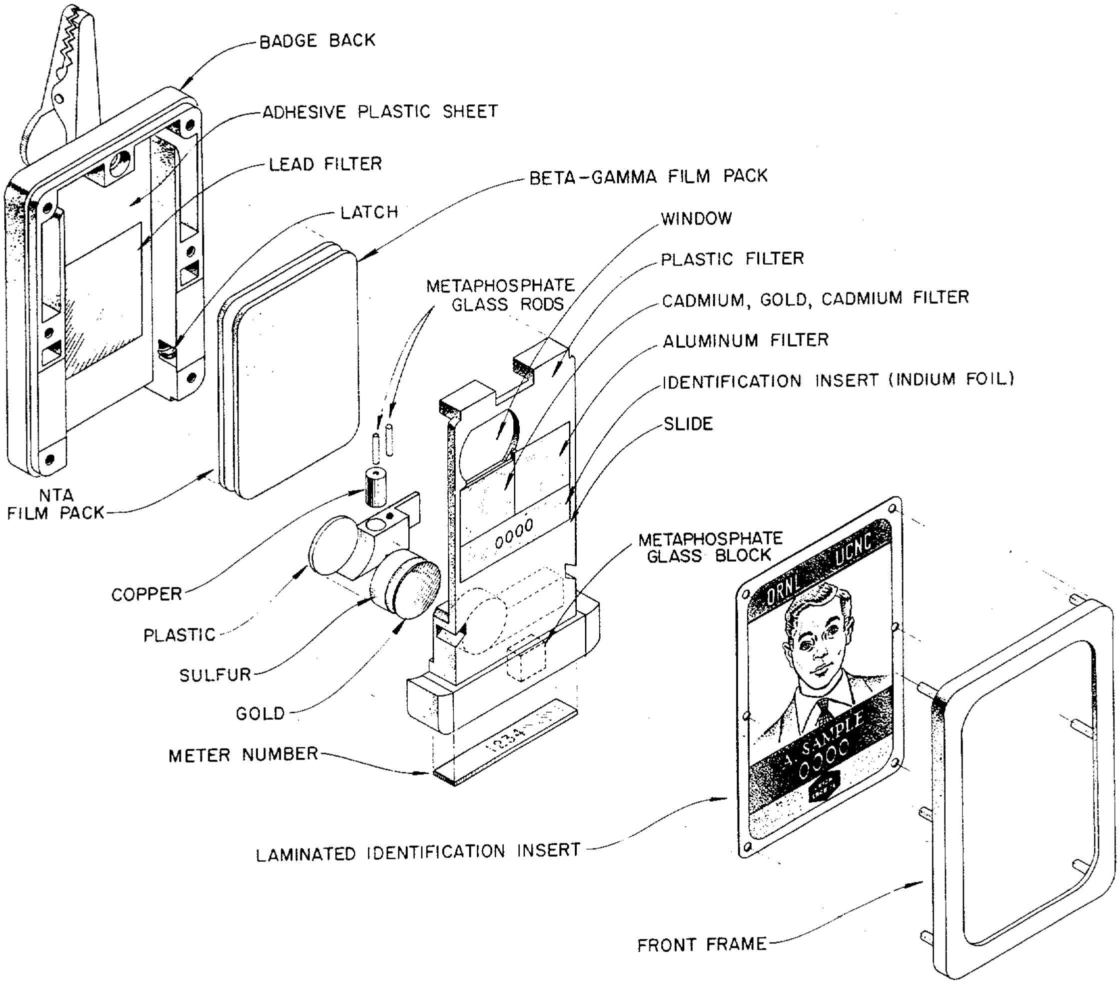

Fig. 6.3 ORNL Badge–Meter, Model II.

Fig. 6.4 Typical Temporary Security Passes Equipped with Monitoring Film.

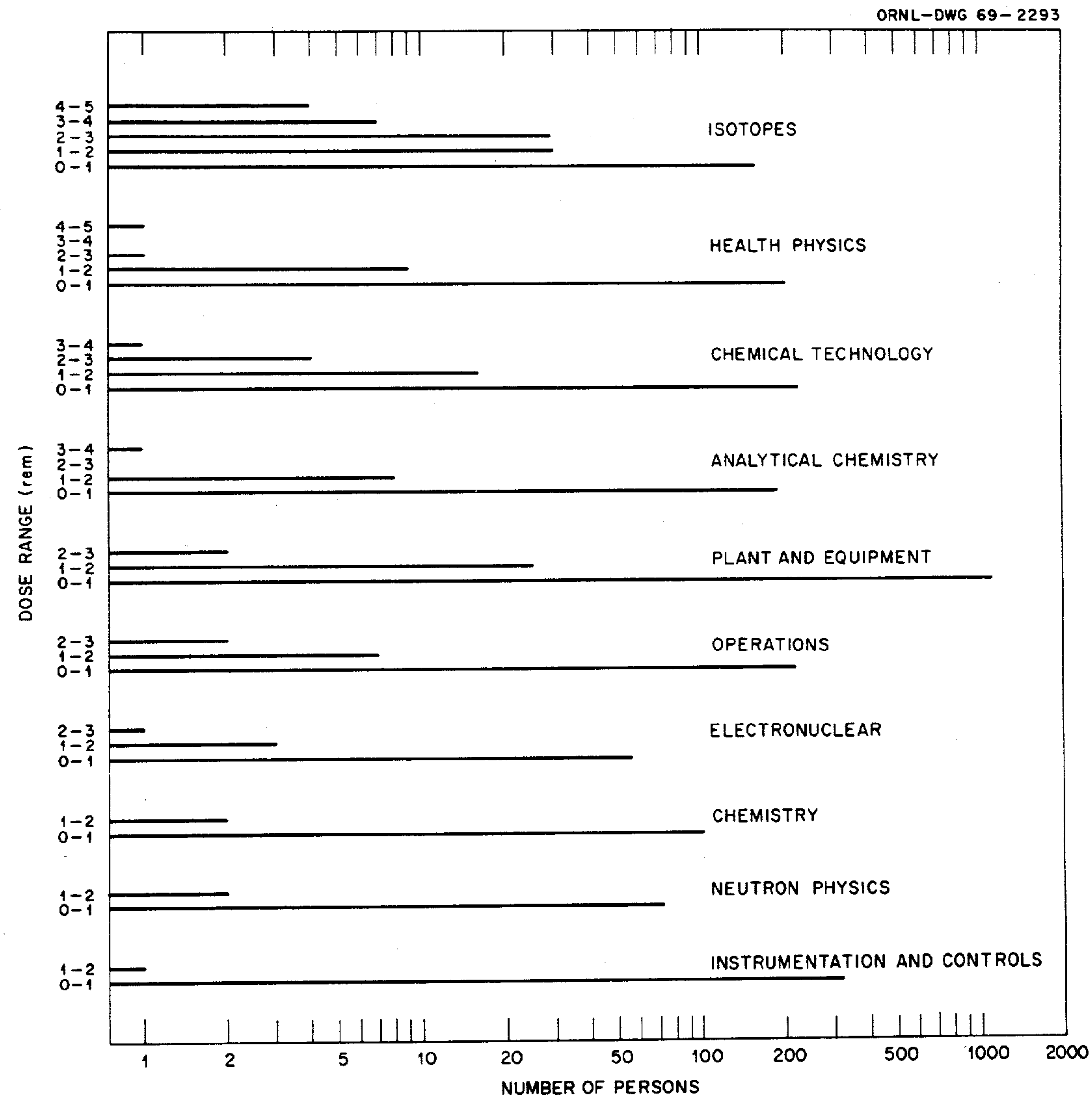

Fig. 6.5 Personnel Dose (Whole Body) by ORNL Division Having One or More Doses, One Rem or Greater, in 1968.

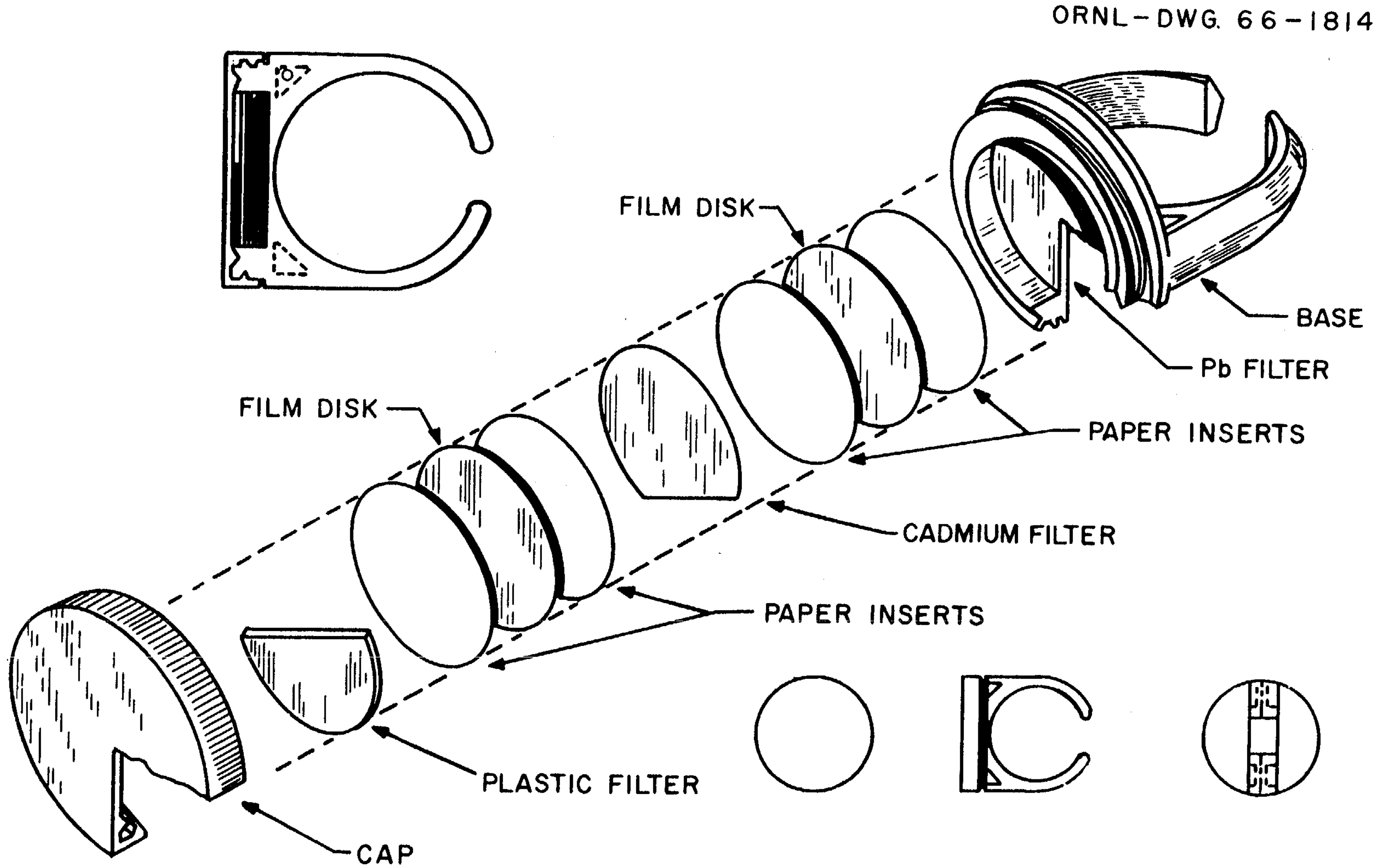

Fig. 6.6 Details of the ORNL Hand Exposure Meter.

ORNL DWG. 6411676

Name	ID Number	Symbol	Dosimetry Dates		Meter Dose	
			Wk-Yr	Qtr-Yr	DS	DC
Last Name, Initials	PR. No.	PF	35-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	31-63	3-63	0.120	0.090
Last Name, Initials	PR. No.	PF	30-63	3-63	0.030	0.000
Last Name, Initials	PR. No.	PF	36-63	3-63	0.070	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	36-63	3-36	0.370	0.310
Last Name, Initials	PR. No.	PF	32-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	33-63	3-63	0.040	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.260	0.130
Last Name, Initials	PR. No.	PF	35-63	3-63	0.040	0.010

Fig. 6.7 Typical ORNL Film Monitoring Data.

ORNL DWG. 64-11675

Name	ID Number	Symbol	Date Wk-Yr	REM DS	REM DC	REM This Qtr DS	REM This Qtr DC	REM This Yr DS	REM This Yr DC	Total REM DC	A	DC/A
----	-----	PF	39-63	0.860	0.630	0.860	0.630	1.68	1.32	35.59	18	2.02
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.340	0.240	0.340	0.240	0.34	0.24	0.24	1	0.80
----	-----	PF	39-63	0.020	0.010	0.020	0.010	0.02	0.01	5.21	14	0.38
----	-----	PF	39-63	0.070	0.040	0.070	0.040	0.30	0.19	18.38	16	1.19
----	-----	PF	39-63	0.390	0.310	0.390	0.310	1.40	1.14	2.74	20	0.14
----	-----	PF	39-63	0.350	0.150	0.350	0.150	0.77	0.49	9.60	17	0.56
----	-----	PEL	27-63	0.010	0.010	0.150	0.120	0.27	0.24	5.55	6	1.09
		PF	39-63	0.140	0.110							
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.400	0.200	0.400	0.200	0.73	0.45	7.43	12	0.64
----	-----	PF	39-63	0.180	0.150	0.180	0.150	0.60	0.49	8.43	7	1.34
----	-----	PF	39-63	0.330	0.110	0.360	0.140	0.81	0.34	3.00	13	0.24
		PN	39-63	0.030	0.030							
----	-----	PF	39-63	0.180	0.080	0.180	0.080	0.51	0.33	29.82	18	1.68
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.320	0.270	0.320	0.270	1.14	0.98	22.76	13	1.76
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.420	0.290	0.420	0.290	1.85	1.11	15.86	16	1.04
----	-----	PF	39-63	0.320	0.140	0.320	0.140	0.67	0.46	8.96	11	0.84
----	-----	PF	39-63	0.390	0.210	0.390	0.210	1.21	0.72	33.62	18	1.87

Fig. 6.8 Typical ORNL Personnel Radiation Exposure Record.

ORNL DWG. 66-4519

DEPT XXXX	HP WK 52												
NAME	PR NO	DC	S	M	T	W	T	F	S	WK	QTR	F	SMB
---------------	-----			0	5	10	0			15	270	62	
---------------	-----	660		10	25		120			155	1140	57	DWQ
---------------	-----										35	11	
---------------	-----										10	1	
---------------	-----			10	10	0	10			30	220	41	
---------------	-----			10						10	115	62	
---------------	-----										5	20	
---------------	-----		0	10	10		20			40	560	54	D Q
---------------	-----										195	22	
---------------	-----											1	
---------------	-----		125	0		0				125	260	60	DW
---------------	-----			30	0	5	0			35	415	61	D
						ENTRIES		+		D	W	Q	COUNT
						23				5	2	2	12

Fig. 6.9 Typical Pocket Meter Weekly Report.

ORNL DWG. 66-4520

RESULTS THIS REPORT 12-20-65

Div. Code	Name	PR NO	HP AREA Number	Type Analysis	Receipt Date	Type Sample	Sample Priority	d/m/Sample	d/m/24 hrs
HP	----------	-----	3550	GAO	12-16-65	U	3		0
HP	----------	-----	3019	PUO	12-12-65	U	3		0
HP	----------	-----	3019	PUO	12-16-65	U	3		0
HP	----------	-----	3019	SRO	12-12-65	U	3		0
Div. Total 4									

Fig. 6.10 Typical Weekly Bio-Assay Sample Status Report.

ORNL-DWG. 69-2039

NAME	ID NO	-------------DIVISION DIV	HP-AREA	DATE	REASON	TYP-SER-NO	RESULTS	REMARKS
------------	------	---	----	07-10-68	WORK AREA	SC-24732	NORMAL HUMAN SPECTRUM	
------------	------	---	----	03-05-68	INCIDENT	CH-24115	NORMAL HUMAN SPECTRUM	
------------	------	---	----	03-05-68	INCIDENT	SC-24116	NORMAL HUMAN SPECTRUM	
------------	------	---	----	05-28-68	WORK AREA	SC-24500	LESS THAN 10PCT MPBB	
------------	------	---	----	01-10-68	FOLLOW-UP	SC-23859	LESS THAN 25PCT MPBB	
------------	------	---	----	02-07-68	FOLLOW-UP	CH-23996	LESS THAN 50PCT MPBB	
------------	------	---	----	02-07-68	FOLLOW-UP	SC-23995	LESS THAN 25PCT MPBB	
------------	------	---	----	03-05-68	INCIDENT	CH-24113	LESS THAN 50PCT MPBB	90SR 30PCT LUNG BURDEN
------------	------	---	----	03-05-68	INCIDENT	SC-24112	LESS THAN 10PCT MPBB	
------------	------	---	----	06-25-68	FOLLOW-UP	SC-24652	LESS THAN 25PCT MPBB	
------------	------	---	----	05-28-68	WORK AREA	SC-24506	NORMAL HUMAN SPECTRUM	

Fig. 6.11 Typical Whole Body Counter (IVGS) Summary Report.

ORNL-DWG. 67-2638

Name - Employee AN

		Symbol	Definition
I.D. Number	5782	DC	Cumulative recorded total rem to whole body since activation date.
S.S. Number	221-16-0038		
Birth Date	6/17/28	DO	Dose data other than that reported herein Yes (3)
Activation Date	1/16/48		

Year	QTR	Rem for Qtr Skin	Rem for Qtr Body	Rem for Year Skin	Rem for Year Body	Rem DC
				DC Prior to 1961		22.13
1961	1	.26	.19			
	2	.20	.16			
	3	.29	.12			
	4	.44	.36			
Total				1.30	.83	22.96
1962	1	.33	.30			
	2*	.56	.48			
	3*	.69	.54			
	4	.59	.51			
Total				2.17	1.83	24.79
1963	1	.61	.50			
	2	.53	.43			
	3	.78	.43			
	4	.03	.03			
Total				1.95	1.39	26.18
1964	1	.04	.03			
	2	.02	.01			
	3	.02	.01			
	4	.09	.04			
Total				.17	.09	26.27
1965	1	.25	.12			
	2	.40	.22			
	3	.48	.28			
	4	.41	.21			
Total				1.54	.83	27.10

Fig. 6.12 Typical Individual External Dose Summary.

7.0 LABORATORY OPERATIONS MONITORING

Radiation incidents are classified according to a severity index system developed over the past several years.[7] The method serves to index unusual occurrences according to degree of severity and permits a system of analysis regarding Health Physics and Safety practices among Laboratory operations. This report summarizes the unusual occurrence frequency rate and discusses some of the problems encountered among Laboratory facilities.

7.1 Unusual Occurrences

During 1968 there were 20 unusual occurrences recorded which represents an increase of 20 percent over the number reported for 1967 (Table 7.1). The number for 1968, twenty, is approximately 23 percent below the five-year average of 26 for the years 1964 through 1968. The frequency rate of unusual occurrences among Laboratory divisions involved (Table 7.2) is known to vary in relationship to the quantity of radioactive material handled, the number of radiation workers involved, and the radiation hazard potential associated with a particular operation or facility.

Thirteen of the incidents reported during 1968 involved area contamination that was handled by the regular work staff without appreciable production or program loss. Eight occurrences involved personnel contamination requiring decontamination under medical supervision, and one incident involved a minor X-ray exposure to the hand of an employee.

7.2 Radiation Surveys

During 1968 Radiation and Safety Surveys personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the established maximum permissible limits. Through seminars, safety meetings and informal discussions with supervision, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of some of the problems and methods of solution.

7.2.1 Health Physics and Safety Assistance during Startup and Test Phase at the ORELA Installation - Health Physics and Safety personnel provided radiation monitoring surveillance during the startup and test phases of ORELA operation. Radiation surveys made during test periods, with the machine operating at approximately 1/50 of its capabilities, indicated the presence of adequate shielding in all areas except those where penetration closures were intentionally left out. Because of the pulsed nature of the beam, the high energy components of the radiation and the presence of electrical fields, evaluation of the stray radiation field surrounding the machine required that conventional dosimetry techniques be supplemented.

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[7] See Applied Health Physics Annual Report for 1963, ORNL-3665, pp. 14-15.

Operation of the machine at full design power is not expected to generate hazardous levels of radiation in areas other than those already designated and identified as Radiation Zones.

7.2.2 Health Physics Coverage at the Radio Isotopic Sand Tracer Project – In a repeat performance of a project instituted during 1967 a representative of the Radiation and Safety Surveys Section acted as a project health physicist at the Radio Isotopic Sand Tracer Tests, conducted by the Technical Services Group of the Isotopes Division for the U. S. Corps of Engineers at Vandenberg Air Force Base, California. The tests involved placing radioactive sand tagged with $^{198-199}Au$ off shore in the ocean and tracing its movement along the ocean floor by the use of a specially designed radiation detection system. The Health Physics representative provided on-the-job surveillance, served as custodian of radioactive material as well as assuring that all Federal and State regulations pertaining to the handling of material were followed. The tests were completed without significant contamination or exposure problems.

7.2.3 Initial Operation of the Carbon Coated ^{233}U-^{232}Th Microspheres Process and the Plutonium Facility in Bldg. 4508 – The process for carbon coating 25 percent ^{233}U – 75 percent Th microspheres was started during the first week in November in the Coated Particle Development Laboratory operated by the Metals and Ceramics Division in Bldg. 4508. These microspheres are to be used in reactor research by Battelle Northwest, Richland, Washington.

Also, the operation of the new plutonium facility in Room 136, Building 4508, was started during the middle of December. This facility is for research and development work on ^{239}plutonium nitride for use in reactors. Health Physics and Safety personnel provided advice and on-the-job surveillance during the initial operational phases of these two processes.

7.2.4 Health Physics Assistance on Specific Projects at Building 3019, Pilot Plant – Health Physics assistance was provided in the planning and monitoring of the Chemical Technology Division's Radiochemical Pilot Plant Operations in Building 3019. One hundred forty-four radiation work permits were certified for a variety of operations involving significant radiation hazards. A brief description of some of the more unusual operations are presented here.

7.2.4(a) Renovation of HRLAF Cell No. 5, Building 3019 – In 1963 containment panels and a polyvinyl chloride pan liner were installed in HRLAF Cell No. 5 to enable analyses of curium samples from Building 4507 with a minimum spread of alpha activity to adjacent cells and cell access areas. In 1968 the cell was restored to its original configuration to provide more flexibility of operation when TRU and TRL supplanted the need of this facility for transuranium analyses. Health Physics helped develop procedures for removal of the grossly alpha contaminated ($> 5 \times 10^5$ α d/m/100 cm^2) panels and material and provided on-the-job monitoring. A plywood enclosure with a "bag-out" port was constructed at the rear opening to the cell. Personnel, working within the enclosure in air supplied plastic suits, placed contaminated materials into large plywood boxes connected by plastic tubing to the "bag-out" port. The boxes were then sealed for transfer to the burial ground. This technique proved very effective for containment of the alpha activity and protection of personnel.

7.2.4(b) Purification, Storage and Redistribution of ^{233}U - Building 3019 at ORNL serves as the national center for distribution of ^{233}U. Solids, which can be dissolved in nitric acid in stainless vessels, and nitrate solutions are received at Building 3019 for storage, purification, and redistribution. One such shipment of nitrate solution containing ~ 9 kg of ^{233}U presented some unique problems in transferring to the purification system in Cell 5. In a number of cases the normal procedure of transferring the bagged bottles of solution into a transfer gloved box could not be followed as some solution had leaked from the primary polyethylene bottles into the secondary plastic bag containers and in some cases into the tertiary steel cylinders spaced at the center of 55 gallon drums. With continuous Health Physics monitoring and after careful planning for criticality safety and contamination control the solution was safely vacuum transferred from the leaking shipping containers into Cell 5 equipment.

7.2.4(c) Preparations of Facilities for Handling and Storage of Plutonium - Facilities for storing up to 100 kg of plutonium as solids (fluroides) or liquid (nitrate) are to be installed in Room 501, Building 3019, adjacent to Pilot Plant Cell No. 7. Special precautions were necessary for protection of personnel and containment of gross alpha contamination ($> 5 \times 10^5$ α d/m/100 cm^2) during the removal of a large solvent recycle tank, piping and other equipment and during excavation of a section of the concrete floor. Walls, ceiling and floor surfaces were thoroughly scrubbed to remove transferable alpha contamination and layers of paint in contrasting colors were applied to bond the remaining relatively fixed alpha contamination. Subsequent air samples indicated concentrations well within MPC_{air} for ^{239}Pu enabling construction personnel to install services in this area with a minimum of restrictions.

7.2.4(d) Procedural Modification for Handling and Containing ^{238}Pu - Sol-gel studies with several hundred gram quantities ^{238}Pu were conducted in the Sol-Gel Pilot Plant, Cell 4, Building 3019. Shielding designed into this facility was adequate for protection of personnel against gamma and neutron radiations. However, procedures normally adequate for containment of ^{239}Pu had to be modified for control of ^{238}Pu due to its higher specific activity and specific heat. Procedure modifications included the use of metal containers, chilled before sealing in plastic bags, and continuous Health Physics monitoring during all bag-out bag-in operations, glove replacements, manipulator repairs and packaging and unpackaging operations. Respiratory protection was also used during these more hazardous operations so that the few minor releases which did occur were of little consequence. A temporary plastic airlock was constructed when a furnace failure necessitated entry into the primary containment cubicle. Personnel wore air supplied plastic suits which were thoroughly washed down before removal. These measures proved adequate for personnel protection and contamination control.

7.2.4(e) Transfer of ^{233}U Oxide to Building 7930 for MSRE Salt Preparation – Eighty-nine cans containing ~36 kg of ^{233}U as oxide were transferred from storage wells in Building 3019 to Building 7930 for MSRE salt preparation. Before loading into the transfer cask, each can was gauged to assure it would fit into the can opener at Building 7930 and a shock absorber was attached. No contamination control problems were encountered, however, careful planning and monitoring was required to minimize external exposure to personnel as readings 2" from the cans ranged to 400 rad/hr. A long handled tool with a suction cup connecting to a vacuum pump was devised to transfer the cans from the storage wells to a gauging device which simultaneously attached a shock absorber. The transfers were completed with no personnel exposures exceeding 20 mrem in any one day.

7.2.5 Operations Involving the Molten Salt Reactor Experiment (MSRE), Building 7503 – The reactor was shut down March 25, after an uninterrupted run of 188 days, and operations were resumed again on September 25. During this scheduled shutdown the ^{235}U fuel was removed from the carrier salt by fluoride volatilization and replaced with ^{233}U fuel prepared at TURF (7930). Various alterations, repairs and inspections were also performed in the system at this time.

Health Physics and Safety personnel assisted in training sessions, and the writing of detailed procedures that aided in the trouble free performance of the various operations. Surface contamination was confined to predetermined zones where it was kept at a low level by periodic cleaning. These were then cleared on completion of the shutdown.

Airborne activity levels were kept well below the $(MPC)_{air}$ for 40-hour week. There were no significant external exposures received by personnel during the performance of the various operations.

7.2.6 Processing the Heavy Elements at the Transuranium Facility, Building 7920 – There was a considerable increase in the amount of heavy elements processed and transferred over the previous year. The following amounts of isotopes were produced during 1968.

Pu	^{243}Am	^{244}Cm	^{249}Bk	^{252}Cf	^{253}Es
17 gms	136 gms	201 gms	1.3 mg	6.8 mg	30 μg

The increased amounts of these isotopes handled plus their unique characteristics (high neutron dose rates, soft gamma and beta radiation in glove boxes and intense alpha concentrations) intensified and extended the problems faced by Health Physics personnel. However, all production schedules were met without significant internal or external exposures to personnel with the average level of exposure about the same as that received during the previous year. Contamination of building surfaces was again kept well below established tolerance values. This is an indication of the excellent cooperation received by Health Physics personnel from among all groups involved in the operation.

7.2.7 Preparation of Enriching Material at TURF (Building 7930) for Use at the MSRE (Building 7503) - Sixty-five kilograms of $^{233}UF_4$ - ^7LiF entectic salt were prepared in Cell "G" as an enriching fuel for the Molten Salt Reactor Experiment. The salt contained 40 kilograms of ^{233}U in mixed oxides of which 225 parts per million were ^{232}U. The hot material necessitated the extensive use of shielding in the operation. Natural uranium was used during the preparation period to simulate actual conditions eliminating the associated hazards of ^{233}U.

The first quantity of ^{233}U was charged into the system on May 9 and delivery to the MSRE, Building 7503, was completed on September 9. The balance of the fuel not required by MSRE was transferred to Cell "B" for storage.

Extensive preplanning and the following of well-developed procedures contributed to adequate control of radiation exposures, contamination and conventional safety hazards, throughout the operation.

7.2.8 Health Physics and Safety Assistance during the Disposal of Liquid Waste at the Shale Fracturing Site - Continuous monitoring was given Operations Division personnel during the injection of $\sim 1.26 \times 10^5$ curies of activity at the Shale Fracturing disposal facility during the year. Radiation exposure to personnel was kept below permissible levels and no significant contamination problems were encountered.

Much of the mixing cell equipment has been replaced with stainless steel material to aid in simplifying decontamination of this equipment after an injection has been completed.

7.2.9 Health Physics and Safety Activities in the Transuranium Research Laboratory during 1968 - Health Physics and Safety personnel continued to collaborate with TRL researchers, working directly with them in planning and conducting specific experiments with transuranium isotopes at the TRL and at other ORNL experimental facilities.

A special series of measurements were made with a 350 µg ^{252}Cf sealed source to determine the relative response of various ORNL radiation survey and personnel monitoring instruments.

Because of the frequent necessity to work at close proximity with intensely radioactive point sources, a thermoluminescent dosimeter reader was purchased and an evaluation of Teflon-LiF dosimeters was begun.

Orientation and training programs in TRL safety requirements and procedures were conducted for ten visiting researchers and numerous groups of Laboratory visitors were given short lecture tours through the facility; additional lectures were given at ORAU and Knoxville College.

7.2.10 The Renovation of the ^{99}Tc Facility in Rooms 9 and 11, Building 3026C - Health Physics assistance was provided during much of the detailed work concerned with the renovation activities in the areas as noted above. Six pre-war hoods (four of which had been modified to serve as glove boxes), a lead shielded solvent extraction cubicle, and two connected tanks, which were located outside the building in metal cabinets under a shed roof, were removed as separate units.

Readings > 10 R/hr under the neutralizer tank were reduced to 80 mR/hr and the inside of the cabinet and contents (tank, lines, etc.) were spray painted. A section of the roof was removed and the cabinets containing the tanks were loaded intact and taken to the burial ground. The glove ports and process lines were sealed and each was removed as a unit to the burial ground. Gram quantities of ^{99}Tc waste were contained in this system and disposal was completed with no significant exposures to personnel and with contamination limited to properly identified zoned areas.

7.3 Laundry Monitoring

A total of 869,863 articles of wearing apparel was monitored at the laundry during 1968. Of the items monitored 3.3 percent were found contaminated.

A total of 12,410 full-face respirators were cleaned and monitored during the year. Of this number, 816 required further decontamination prior to being placed back in service.

Of the 424,774 khaki garments monitored during the year, 118 were found contaminated. This is an increase of about 50 percent from last year.

During the year a filtered exhaust hood was installed in the laundry monitoring room for use during the monitoring of all laundry and during the separation and handling of all contaminated laundry. The installation of this equipment lends assurance that the wide-spread dispersal of dust material from the clothing has been eliminated.

Table 7.1 Unusual Occurrences Summarized for the 5-Year Period Ending with 1968

	1964	1965	1966	1967	1968
Number of Unusual Occurrences Recorded	29	41	22	16	20
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no cleanup effort	14	11	8	5	7
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamination	15	30	14	11	13
1. Personnel Exposures	9	12	8	5	9
a. Nonreportable overexposures with minor work restrictions imposed	9	11	8	5	9
b. Reportable overexposures with work restrictions imposed	0	1	0	0	0
2. Contamination of Work Area	15	28	14	11	13
a. Contamination that could be handled by the regular work staff with no appreciable departmental program loss	14	27	12	11	13
b. Required interdepartmental assistance with minor departmental program loss	1	1	2	0	0
c. Resulted in halting or temporarily deterring parts of the Laboratory program	0	0	0	0	0

Table 7.2 Unusual Occurrence Frequency Rate within the Divisions for the 5-Year Period Ending with 1968

Division	No. of Unusual Occurrences					5-Year Total	Percent Lab. Total (5-Year Period)
	1964	1965	1966	1967	1968		
Analytical Chemistry	3	6	1	3	4	17	13.2
Biology		1			1	2	1.6
Chemical Technology	3	8	3	4	5	23	18.0
Plant and Equipment	2	2	2		1	7	5.5
Inspection Engineering	1					1	.8
Electronuclear Research		1	1			2	1.6
Health Physics	1	2				3	2.3
Instrumentation and Controls	1					1	.8
Isotopes	12	10	8	4	6	40	31.2
Metals and Ceramics				1	1	2	1.6
Neutron Physics					2	2	1.6
Operations	3	8	4			15	11.7
Physics	3	2	1	1		7	5.5
Reactor			2	3		5	3.8
Reactor Chemistry		1				1	.8
TOTALS	29	41	22	16	20	128	100.0

8.0 INDUSTRIAL SAFETY

The safety record for 1968 was the best in the history of the Laboratory. ORNL experienced only one Disabling Injury during the year. There were fewer Serious Injuries reported this year (1968) than for any year for the past five years. The number of medical treatment cases for 1968 also showed a marked decline as compared with 1967.

8.1 Accident Analyses

The Disabling Injury Frequency Rate for 1968 was 0.13. The average frequency rate for the previous five years, 1963–1967, was 1.19. The Disabling Injury history of the Laboratory for the five-year period 1964 through 1968 is shown in Table 8.1. The Disabling Injury frequency rates since the inception of Union Carbide as the contractor at ORNL are shown in Figure 8.1.

There were 12 divisions which did not have a Serious or Disabling Injury during 1968. There are 17 divisions which have accumulated 1,000,000 or more hours since the last Disabling Injury. The Serious Injury, Disabling Injury, and exposure-hour data for ORNL divisions are shown in Table 8.2.

Table 8.3 includes injury data for the four facilities—ORNL, Paducah, Y-12, and ORGDP. The frequency rates for Disabling Injuries for three of the four Carbide facilities decreased in 1968 as compared with 1967. The frequency rates for Serious Injuries also decreased at three of the facilities, including ORNL. Serious Injuries at ORNL decreased from 93 in 1966, to 89 in 1967, to 72 in 1968. The frequency rates for Disabling Injuries and Serious Injuries at ORNL for the past five years, 1964–1968, are shown graphically in Figure 8.2.

There were 1,260 injuries (includes first aid, Serious Injuries, and Disabling Injuries) reported during 1968. Figures 8.3, 8.4, and 8.5 show injury data according to type of accident, the nature of the injury, and the part of body injured.

8.2 Analysis of Disabling Injury

The following is a brief analysis of the Disabling Injury experienced at ORNL July 3, 1968.

A machinist was chucking a piece of polished steel in a 36" Lehmann hydraulic lathe. Aluminum spacers were being placed between the steel and the jaws to prevent scratches. While the chuck was being rotated to a position to insert the fourth and final spacer, the tips of the man's third and fourth fingers of his left hand were pinched between the chuck wrench and the lathe apron, amputating a part of the first joint of the fourth finger. The chuck wrench, a heavy crank, was bent when it struck the apron, indicating that the chuck was rotating under power (speed was set a 5 rpm).

8.3 Safety Award Periods - 1968

ORNL had accumulated more than five million man-hours without a Disabling Injury before the injury which occurred on July 3, 1968, ended the award period. As the Laboratory did not experience a Disabling Injury for the remainder of the year, and, as the Safety Awards Plan will be replaced with a new Safety Incentive Plan effective January 1, 1969, the first three million Disabling Injury-free hours following the July 3 incident were considered in determining the award value for 1968. Thus, the value of the awards were $15.00 ($10.00 for five million man-hours; $5.00 for three million man-hours).

Table 8.1 Disabling Injury History – ORNL, 1968

	1964	1965	1966	1967	1968
Number of Injuries	8	18	4	4	1
Labor Hours (Millions)	7.5	7.7	7.8	8.0	7.8
Frequency Rate	1.07	2.34	0.51	0.50	0.13
Days Lost or Charged	1107	2816	231	245	60
Severity Rate	148	366	30	31	8

Table 8.2 Injury Record by Divisions - 1968

Division	Medical Treatment Cases	Number of Serious Injuries	Disabling Injuries Number	Freq.	Sev.	Exposure Hours	Exposure Hours Since Last Disabling Injury
Analytical Chemistry	21	3				325,579	5,444,655
Chemical Technology	35	4				448,916	4,363,296
Chemistry	15					201,508	741,731
Director's	7					234,420	1,042,444
Electronuclear	2					108,763	195,504
Health Physics	35	2				380,580	1,813,047
Instr. & Controls	51	1				582,278	2,108,833
Mathematics	5					179,487	1,514,306
Metals & Ceramics	47	5				591,330	2,873,999
Neutron Physics	6	1				146,025	1,061,304
Physics	8					110,857	2,635,989
Reactor	2					96,026	1,181,565
Reactor Chemistry	11					190,303	1,597,888
Solid State	3	1				157,995	1,982,028
General Engineering	10					343,418	1,704,249
Health	2					57,494	948,388
Isotopes	34	1				292,126	1,001,884
Laboratory Protection	17					143,679	339,208
Operations	64	4				442,522	1,470,841
Personnel	16	3				200,776	250,580
Plant & Equipment	853	47	1	0.46	28	2,155,832	1,013,657
Technical Information	12					241,055	2,259,500
Inspection Engineering	4					88,241	693,610
PLANT TOTAL	1260	72	1	0.13	8	7,719,210	3,675,171

Table 8.3 Four-Plant Tabulation of Injuries - 1968

	Labor Hours (Millions)	Disabling Number of Injuries	Disabling Frequency Rate	Days Lost or Charged	Severity Rate	Serious Number of Injuries*	Serious Frequency Rate
ORNL	7.8	1	0.13	60	8	72	9.3
ORGDP	4.9	0	0.0	0	0	48	9.8
Y-12	11.4	2	0.18	227	20	116	10.2
Paducah	2.1	3	1.42	322	153	29	13.8

*Includes the number of Disabling Injuries.

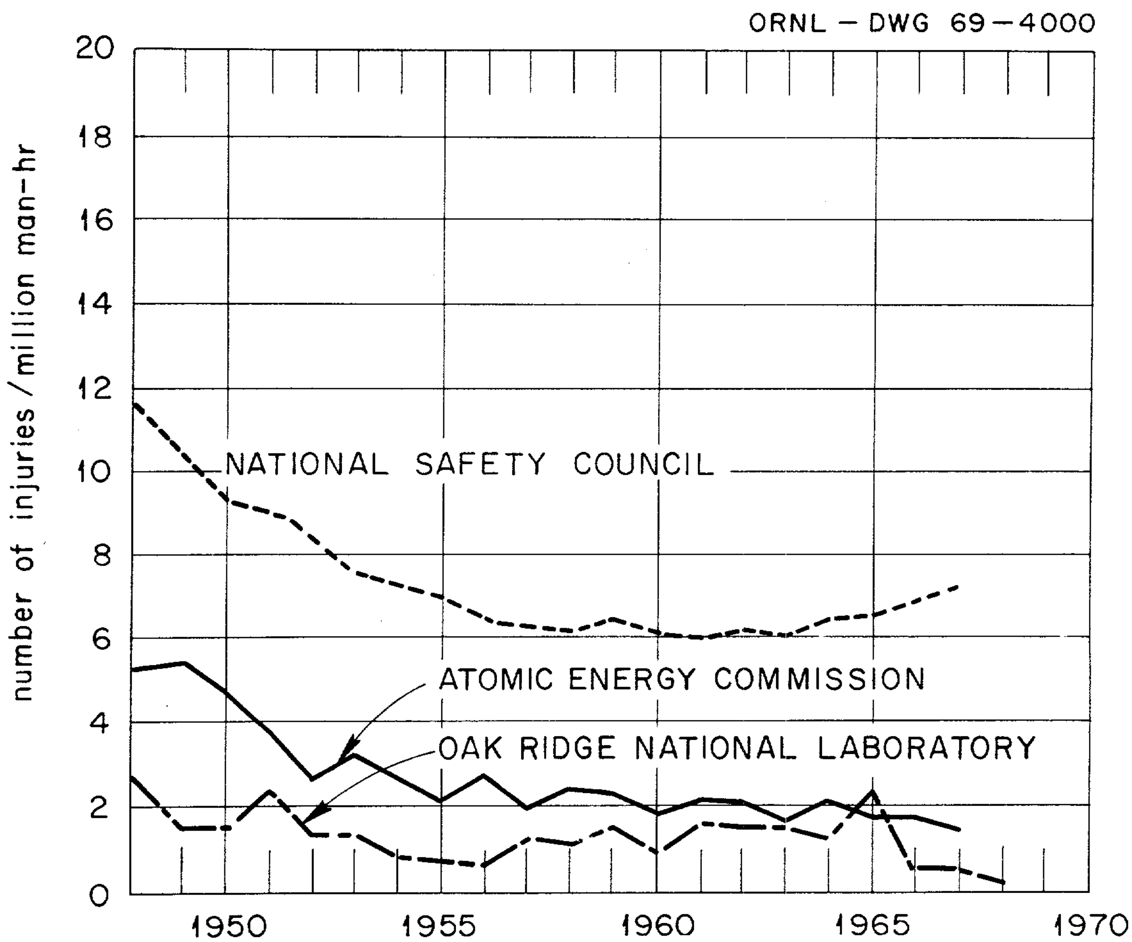


Fig. 8.1 Disabling Injury Frequency Rates Since Inception of Carbide Contract Compared with Frequency Rates for AEC-NSC.

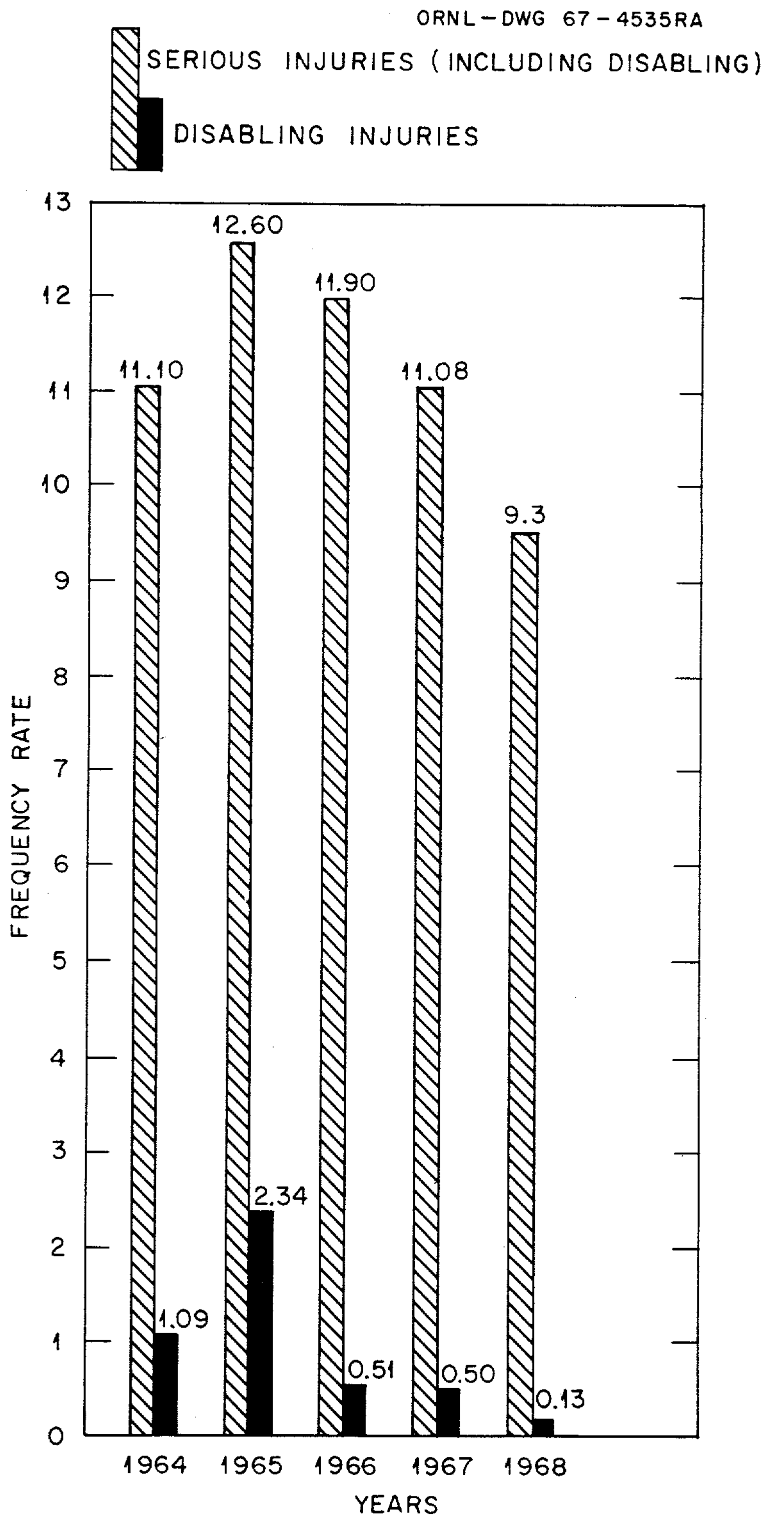


Fig. 8.2 Injury Frequency Rates—1964-1968.

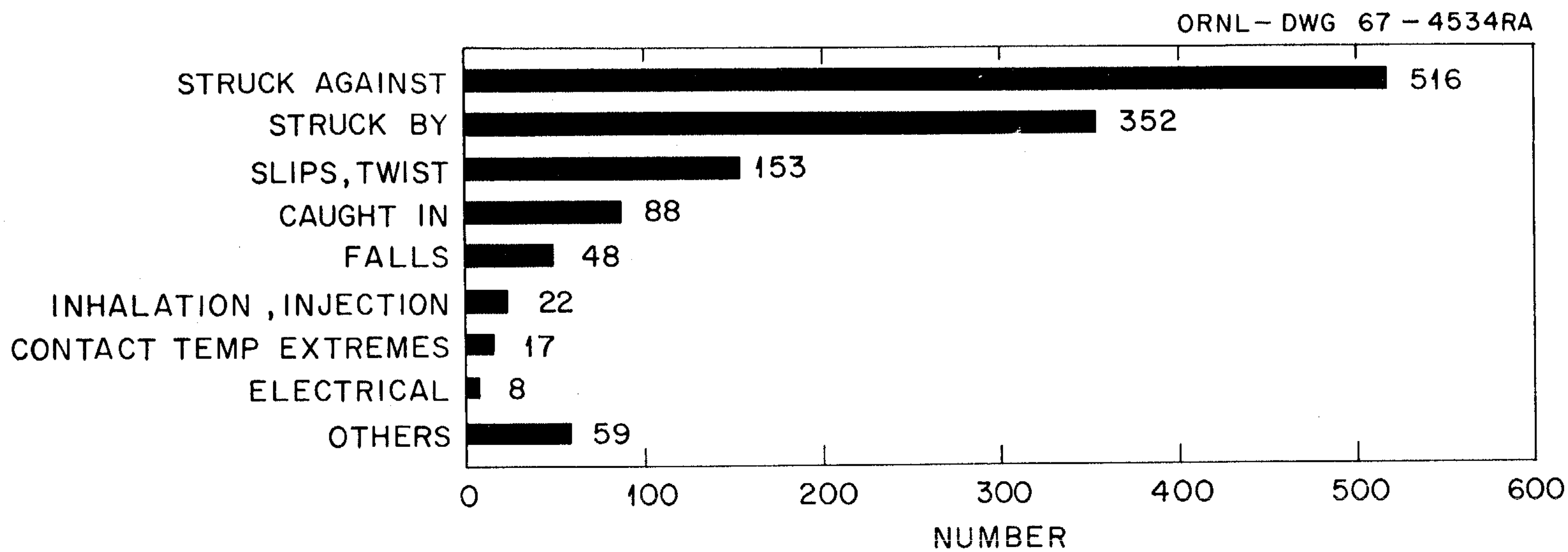


Fig. 8.3 Accidents by Types

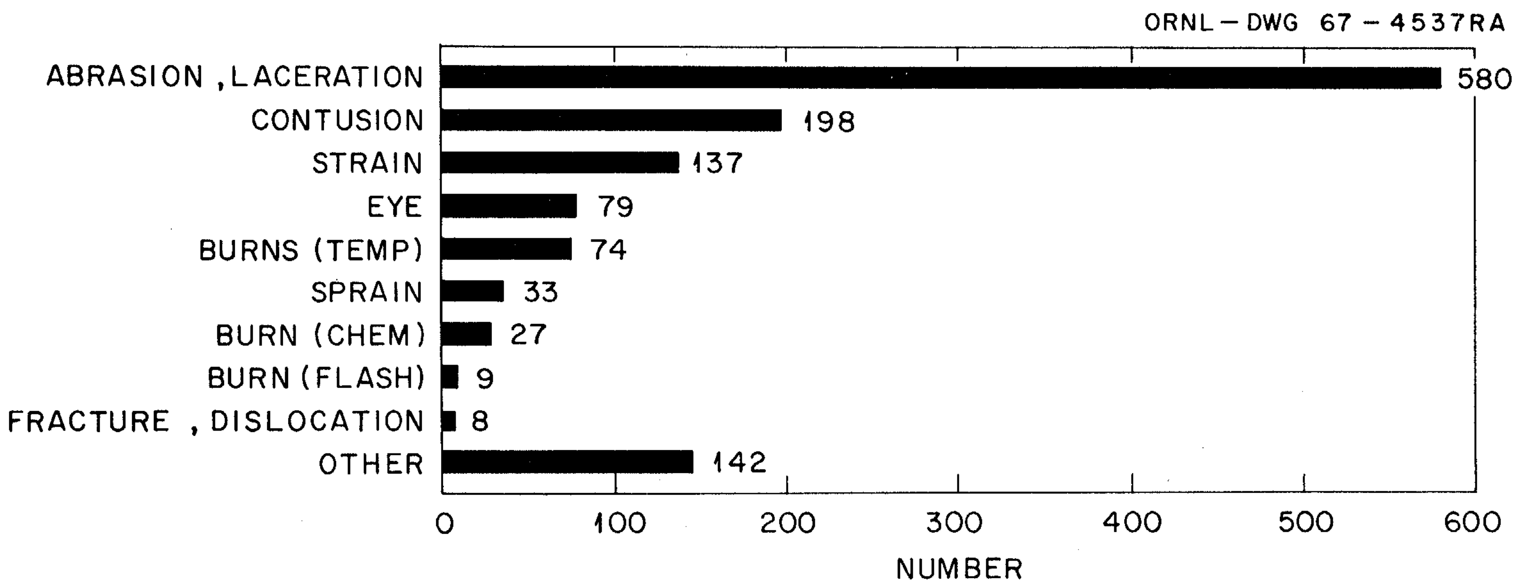
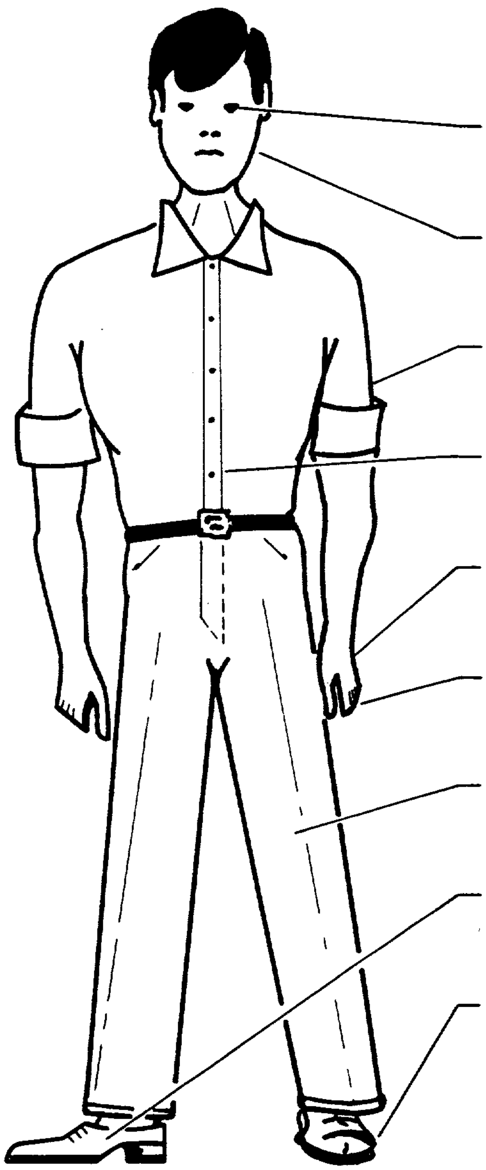


Fig. 8.4 Nature of Injury

ORNL – DWG 69 – 3999



BODY AREA	%	TOTAL INJURIES
EYES	7.6	92
HEAD	6.7	80
ARMS	10.8	132
TRUNK	4.9	60
HANDS	13.3	160
FINGERS	42.0	510
LEGS	7.7	94
FEET	4.4	52
TOES	0.3	3
GENERAL	2.3	28

Fig. 8.5 Part of Body Injured

9.0 LABORATORY ASSAYS

Laboratory Assays Units provide laboratory support to the Health Physics Monitoring Sections. These services include (1) the analysis of body fluids and excreta (bio-assay) for the monitoring of personnel for internal radiation exposure, (2) the radiochemical analysis of environs samples, (3) counting services for the environs monitoring and radiation survey programs, (4) autoradiography, and (5) whole body counting (in vivo gamma spectrometry).

9.1 Bio-Assay Analysis

The number and types of analyses performed by the Bio-Assay Unit during 1968 are given in Table 9.1. A total of 5120 analyses were performed which include 4775 analyses on samples submitted by donors and 345 analyses on standard and blank samples analyzed for control purposes. Approximately 79 percent of the samples were analyzed for either the alpha emitters or strontium.

9.2 Counting Facility

The counting facility processed 197,555 samples during 1968. A tabulation of the number and types of samples counted is presented in Table 9.2. The total number of samples processed was about the same as the previous year.

9.3 Environs Monitoring Sample Analysis

Table 9.3 presents the number and type of environs samples analyzed and the type of analysis performed on each type of sample. A total of 5,274 samples was analyzed during 1968 as compared with 5,995 samples analyzed in 1967. Analysis of environs monitoring samples may range from a single determination to as many as 12 determinations per sample depending upon the radionuclides present. The methods used by the various analytical groups are generally described in the ORNL Master Analytical Manual.

9.4 Autoradiography

There were 1,848 films processed during 1968 in support of radioparticulate studies conducted by the Environs Monitoring Units.[8]

9.5 Whole Body Counter

During the calendar year 1968 the whole body counting program included 964 counts on 810 persons; 821 or approximately 86 percent of these counts showed a

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[8] Methods described in ORNL-2601, Radioactive Waste Management at Oak Ridge National Laboratory.

normal human spectrum. Of the 964 counts, 65 were initial counts made on persons involved in possible contamination incidents. Of the 65 counts, 35 showed evidence of internal contamination present. Sixty-seven counts were made for the purpose of further investigation of positive counts; all but five of these showed evidence of continued internal contamination.

In addition to the whole body counts noted above, two counts made on the wound probe following minor injuries in contaminated areas showed positive evidence of contaminated wounds. Seventy-five counts were made for calibration or standard counts, 65 counts were made to identify contamination in air or surface contamination samples, seven counts were made in an attempt to pinpoint the source of unexplained fluctuations in the natural background in the iron room and 23 counts were made for the purposes of developing and improving in vivo or wound probe counting capabilities.

There was no case, based on data collected by the IVGS, for which the AEC reportable level for occupational workers (one-half of a permissible body burden averaged over the year) was exceeded.

Table 9.1 Bio-Assays Analyses - 1968

Analytical Procedure	Number of Analyses	
Urine:		
Gross α	22	
Trans Pu	469	
Sr	1,262	
U	594	
TRE (total rare earths)	1	
^{3}H	134	
^{137}Cs	198	
^{239}Pu	1,373	
Np	20	
Gross β	3	
Other	663	
Total		4,739
Fecal:		
Pa	6	
Gross α	2	
Sr	3	
U	2	
Trans Pu	10	
Others	13	
Total		36
Standards and blanks		345
GRAND TOTAL		5,120

Table 9.2 Counting Facility Resumé--1968

Types of Samples	Number of Samples			Unit Total	Weekly Average
	Alpha	Beta	Gamma		
Survey Area Samples					
Smears	73,736	71,415		145,151	2,791.4
Air Filters	22,086	22,086		44,172	849.5
Environs Monitoring					
Air Filters	1,822	1,822		3,644	70.1
Gummed Paper		1,472		1,472	28.3
Rain Water		1,416		1,416	27.2
White Oak Lake Effluent	123	1,068		1,191	22.9
Animal Thyroids			62	62	1.2
Milk			447	447	8.6
GRAND TOTAL	97,767	99,279	509	297,555	3,799.1

Table 9.3 Environmental Monitoring Samples--1968

	Sample Type	Type of Analyses	Number Samples
1.	Monitoring network filters	Gross beta, autoradiogram	1,827
2.	Gummed paper fallout trays	Gross beta, autoradiogram	1,502
3.	Rain water	Gross beta	720
4.	White Oak Dam effluent	Gross beta, radiochemical, gamma spectrometry	481
5.	Clinch River water	Gross beta, radiochemical, gamma spectrometry	20
6.	Raw milk	Radiochemical	469
7.	Pasture grass	Radiochemical, gamma spectrometry	185
8.	Potable water	Radiochemical, gamma spectrometry	8
9.	Silt composites	Radiochemical, gamma spectrometry	0
10.	Animal thyroids*	Gamma spectrometry	62
		TOTAL	5,274

*Discontinued 4-3-68.

10.0 HEALTH PHYSICS INSTRUMENTATION

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the selection of electronic radiation monitoring instruments used in the Laboratory health physics program. Normally, the Health Physics Division is responsible for determing the need for new instrument types and modifications to existing types, for specifying the health physics requirements and for approval of the design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or cross-ordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated, and maintained by Health Physics Division personnel.

10.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are AC powered. Portable instruments are assigned and issued to the Radiation Survey Units. Stationary instruments are the property of the Laboratory division which has the monitoring responsibility in the area in which the instrument is located. Table 10.1 lists portable instruments assigned at the end of 1968; Table 10.2 lists stationary instruments at the X-10 site in use at the end of 1968. There were net increases in 1968 of 25 portable instruments and 25 stationary instruments.

During 1968, 400 new pocket meters, 502 new fiber dosimeters (200 mR range) and 171 personal radiation monitors (PRM) were issued by ORNL Stores. Most of the pocket meters issued were replacements for instruments which had been lost or damaged.

Inventory and Service Summaries for health physics instruments are prepared on a CDC 1604. These computer programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments at the X-10 site by division is shown in Table 10.3.

10.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel.

Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics instruments.

Portable instruments should be serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for 1968 is shown in Table 10.4.

Stationary instruments are calibrated by Calibrations Group personnel or by Radiation and Safety Surveys personnel who use sources which are designed, standardized, and provided by the Calibrations Group.

10.3 Instrumentation Developments and Innovations

10.3.1 Beta Dosimeter - The response of field-type radiation survey instruments to beta radiation from small-area sources has been determined. Observational readings were made with typical instruments from beta sources which were prepared for the purpose. The dose rates to the skin from such sources were derived from measurements and calculations.

It was found from these experiments that an adequate estimate of skin dose rates might be difficult to obtain with most field-type instruments.

The thin (2 mg/cm^2) end window GM Counter (Amperex 250RC) had a response which was relatively high and almost directly proportional to the absorbed dose rate at all energies which were used in the study (ref. Table 10.5).

A transistorized, portable GM Survey Meter (Model ORNL-Q-2826) was developed especially for use with the 250RC counter. The Q-2826 has three ranges: 500, 5000 and 50,000 counts per minute, and it will continue to read off-scale if the full scale counting rate is exceeded.

The counter is equipped with an aluminum collimator which provides for defining the area observed and the distance of the counter from the skin. The instrument as used with the collimator (Figure 10.1) is calibrated in terms of mrad per hour averaged over an area of one square centimeter.

10.3.2 NAD Test - The Radiation Dosimetry Section participated in the 1968 Nuclear Accident Dosimetry Intercomparison conducted at the DOSAR Facility during the period July 8 - 19, 1968.

Table 10.1 Portable Instrument Inventory - 1968

Instrument Type	Instruments Added 1968	Instruments Retired 1968	Assigned Inventory Jan. 1, 1969
GM Survey Meter	11	1	462
Cutie Pie	5	3	462
Juno	0	4	28
Alpha Survey Meter	19	0	256
Neutron Survey Meter	0	0	88
Miscellaneous	0	2	13
TOTAL	35	10	1309

Table 10.2 Inventory of Facility Radiation Monitoring Instruments for the Year - 1968

Instrument Type	Installed During 1968	Retired During 1968	Total Jan. 1, 1969
Air Monitor, Alpha	3	1	93
Air Monitor, Beta	2	3	180
Hand-Foot Monitor	0	0	32
Lab Monitor, Alpha	4	2	147
Lab Monitor, Beta	8	2	211
Monitron	4	0	238
Other	18	6	246
TOTAL	39	14	1147

Table 10.3 Health Physics Facility Monitoring Instruments
Divisional Allocation at X-10 Site - 1968

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	4	12	11	16	15	6	64
Chemical Technology	47	49	53	35	38	36	258
Chemistry	9	9	19	24	20	8	89
Metals and Ceramics	12	16	16	17	11	12	84
Isotopes	12	30	19	41	53	17	172
Operations	1	41	2	19	54	14	131
All Others	8	23	27	59	47	185	349
TOTAL	93	180	147	211	238	278	1147

Table 10.4 Calibrations Resumé - 1968

		1967	1968
A.	Portable Instruments Calibrated		
1.	Beta-Gamma	3,877	3,755
2.	Neutron	203	229
3.	Alpha	1,063	1,100
4.	Pocket Chambers and Dosimeters	3,815	1,779
B.	Films Calibrated		
1.	Beta-Gamma	1,684	1,598
2.	Neutron	40	20

Table 10.5

Sources on Skin

Dose Rates to Skin Compared with Readings
Taken with Instruments at 1 cm from the Sources

Source (0.1 μCi)	Skin Dose Rate mrad/hr	Instrument Reading				
		Juno mrad/hr	V-440 mrad/hr	1B85 cts/min	106C cts/min	250RC cts/min
^{32}P	200	2.6	2.8	6000	5100	22,000
^{89}Sr	200	2.7	2.9	5500	4000	22,500
^{111}Ag	190	2.5	2.6	3700	2500	20,500
^{36}Cl	190	2.5	2.7	3800	1500	18,000
^{185}W	170	2.4	2.6	1100	300	15,000
^{60}Co*	150	2.4	2.4	500	350	14,000
^{147}Pm	110	2.1	2.1	15	< 10	11,500

*γ emitter, also.

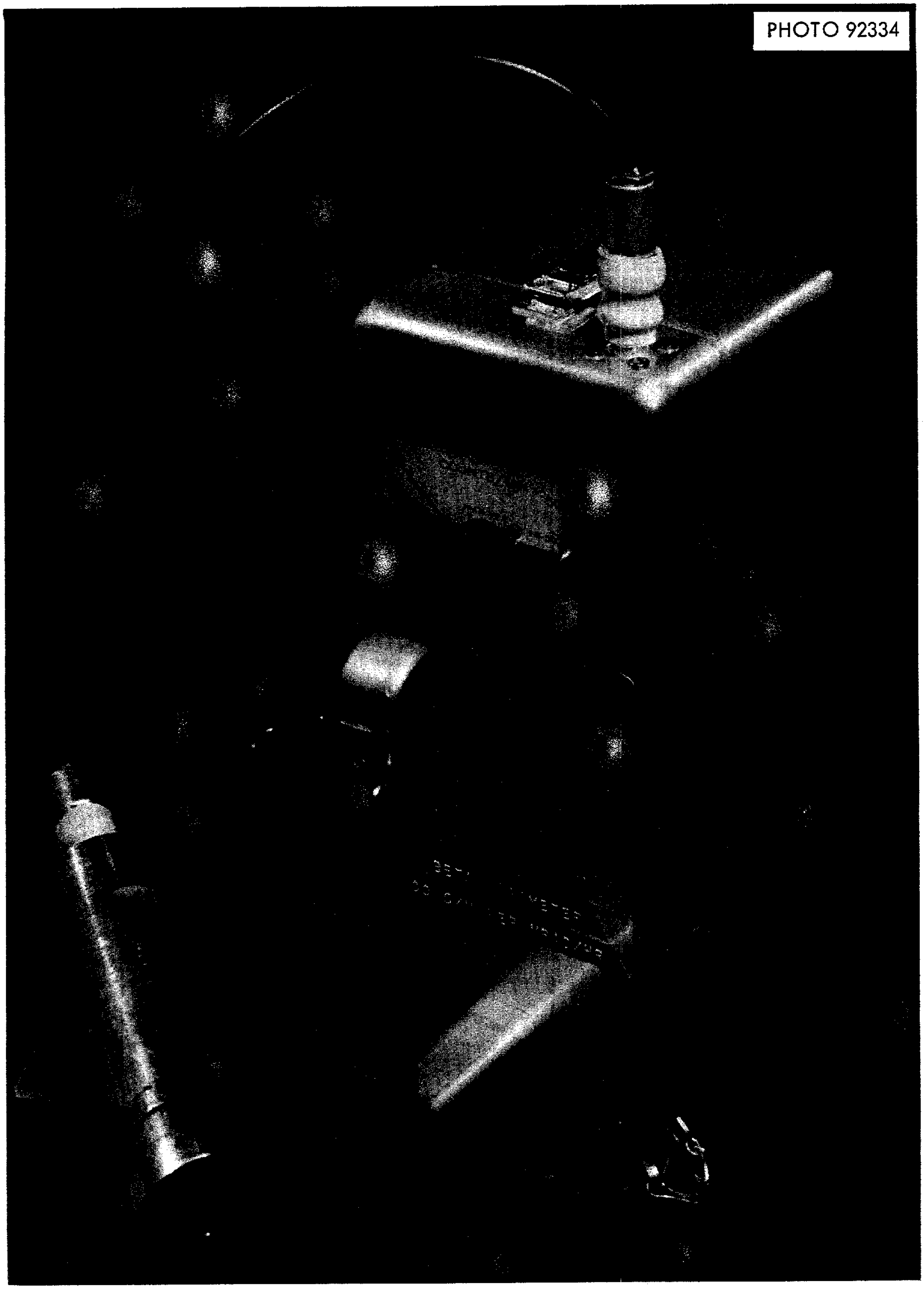

Fig. 10.1 Beta Dosimeter

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D. M. Davis, Health Physics and Safety Annual Report for 1967, ORNL-4286, August, 1968.

E. D. Gupton, "Alpha Air Monitor for ^{239}Pu", paper presented at the Thirteenth Annual Meeting of the Health Physics Society, Denver, Colorado, June 17-20, 1968.

E. D. Gupton, "Methods and Procedures for External Radiation Dosimetry at ORNL", January 1, 1968, ORNL CF 68-1-73.

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W. S. Snyder, B. R. Fish, S. R. Bernard, M. R. Ford, and J. R. Muir, "Urinary Excretion of Tritium Following Exposure of Man to HTO—A Two Exponential Model", Phys. Med. Biol., 1968, Vol. 13, No. 4, 547-559.

12.0 VISITORS AND TRAINING GROUPS

During 1968 there were 91 visitors to Health Physics and Safety, as individuals or in groups, for training purposes. This figure represents a decrease of ~ 14 percent from the number of visitors recorded during 1967. Table 12.1 is a listing of training groups which consisted of five or more persons.

Table 12.1 Training Groups in Health Physics and Safety Facilities during 1968

Facility	Number	Training Period
Youngstown State University	11	3/19/68 - 3/19/68
University of Arkansas (Radiological Health)	5	4/23/68 - 4/26/68
ORAU Health Physics Institute	12	5/27/68 - 5/30/68
AEC Fellowship	5	6/17/68 - 6/17/68
ORAU Students	16	10/28/68 - 11/1/68

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